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Shutdown Decay Heat Removal Analysis of a Babcock and Wilcox Pressurized Water Reactor

Case Study

Prepared by W. R. Cramond, D. M. Ericson, Jr., G. A. Sanders

Sandia National Laboratories

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Case Study

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ABSTRACT

This is one of six case studies for USI A-45 Decay Heat Removal (DHR) Requirements. The purpose of this study is to identify any potential vulnerabilities in the DHR systems of a typical Babcock and Wilcox PWR, to suggest possible modifications to improve the DHR capability, and to assess the value and impact of the most promising alternatives to the existing DHR systems. The systems analysis considered small LOCAs and transient internal initiating events, and seismic, fire, extreme wind, internal and external flood, and lightning external events. A full-scale systems analysis was performed with detailed fault trees and event trees including support system dependencies. The system analysis results were extrapolated into release categories using applicable past PRA phenomenological results and improved containment failure mode probabilities. Public consequences were estimated using site specific CRAC2 calculations. The Value-Impact (VI) analysis of possible alternatives considered both onsite and offsite impacts arriving at several risk measures such as averted population dose out to a 50-mile radius and dollars per person rem averted. Uncertainties in the VI analysis are discussed and the issues of feed and bleed and secondary blowdown are analyzed.

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1.0 INTRODUCTION

1.1 Objectives

The objectives of Task Action Plan A-45 are to evaluate the safety adequacy of decay heat removal (DHR) systems in existing light water reactor nuclear power plants and to assess the value and impact (benefit-cost) of alternative measures for improving the overall reliability of the DHR function if required. To meet these objectives a program was developed which assesses the adequacy of the shutdown portion of the DHR system reliability in existing plants, establishes the feasibility of potential measures to improve DHR, and provides a value/impact analysis of the most promising of such measures.

1.2 Background

Because Unresolved Safety Issue A-45, Decay Heat Removal Requirements encompasses the entire industry, the population of nuclear power plants to be considered for analysis started with approximately 173 units. Seventy units were excluded from study because: 1) they were of special types (e.g., HTGR) or included in the Safety Evaluation Program, 2) there were sufficiently similar units included in the remaining base, or 3) the units were not far enough along in the design/construction process to have information readily available. The remaining 90 plus units formed the base of the TAP A-45 study.

The initial steps in assessing the adequacy of DHR were to: 1) characterize the units in terms of their physical parameters, i.e., number and location of safety pumps, number of redundant emergency power trains, etc., and 2) develop a set of qualitative screening questions against which the plant characteristics could be compared. This set of qualitative screening questions was based upon a thorough review of existing guidance such as the Standard Review Plan and the various Regulatory Guides, previous Probabilistic Risk Assessments, special topical studies such as the Auxiliary Feedwater Studies, etc. The intent was to establish a set of questions which would reveal potential deficiencies in DHR capabilities for both Design Basis Events and for beyond Design Basis Situations. After review and reduction, some 135 questions remained for use in the qualitative screening process. There are four possible outcomes of such a qualitative screening: the plant meets the postulated condition; the plant does not meet the postulated condition; there is insufficient plant information in the data base to evaluate the design; or the question does not apply to that plant. It must be remembered that the sole purpose of this qualitative screening was to highlight potential DHR vulnerabilities for further study. Because each plant has a unique configuration, an inability to meet the conditions of a particular question at one plant may be more or less serious than a similar inability at another plant. For example,

consider the case of a single valve in a DHR water supply line. There is a screening question designed to identify such valves which, if closed, would fail the system. If this valve is in the line from the only source of water, it may be a potentially serious single failure mode. However, if there exist redundant sources of water, redundant lines, or other systems which perform the same function, the single valve may be of little concern. Such differences can only be established by detailed analysis. Nevertheless, the qualitative screening serves as a tool to focus program resources and direction.

On the basis of the initial qualitative screening, approximately twenty plants, which included all vendor types, were potential candidates for further study. From this group seven plants were selected. This selection process was subjective and took into account other issues in which a particular plant might be involved, operational status, utility willingness to participate, and similar concerns. The ANO-1 Nuclear Plant was one of those selected for more detailed study.

1.3 The ANO-1 Nuclear Plant

ANO-1 is one of two units on the Arkansas Nuclear One site. Unit one is a Babcock and Wilcox closed cycle pressurized water nuclear steam supply system. Unit two is a Combustion Engineering unit essentially independent of unit one and is not addressed in this study.

ANO-1 is designed to operate at core power levels up to 2568 Mwt with a net output of 836 MWe. It uses chemical shims and control rods for reactivity control and generates steam with a small amount of superheat in once-through steam generators. The reactor building is a fully continuous reinforced concrete structure in the shape of a cylinder on a flat foundation slab with a shallow domed roof. The cylindrical portion is pre-stressed by a post tensioning system consisting of horizontal and vertical tendons. The foundation slab is reinforced with high-strength reinforcing steel. A welded steel liner is attached to the inside face of the concrete shell.

The reactor building completely encloses the entire reactor and reactor coolant system including the steam generators and portions of the engineered safeguards systems. The auxiliary building houses the safety related systems including the high and low pressure safety injection, reactor building spray system, reactor building cooling system, emergency feedwater system, emergency diesel generators, electrical switchgear rooms, battery rooms, and the control room. The turbine building encloses the unit one and unit two generators and related equipment. The condensate storage tanks and borated water storage tank are located outside west of the auxiliary building.

Several aspects of the plant and equipment layout are presented in more detail in Figures 1.1 to 1.6. Four levels are shown: below grade elevation 335', ground floor 354', intermediate level 372', and operating level 385'. Various DHR related rooms and equipment are shown. Plant design features are discussed in the special emergency analyses, especially Appendix I.

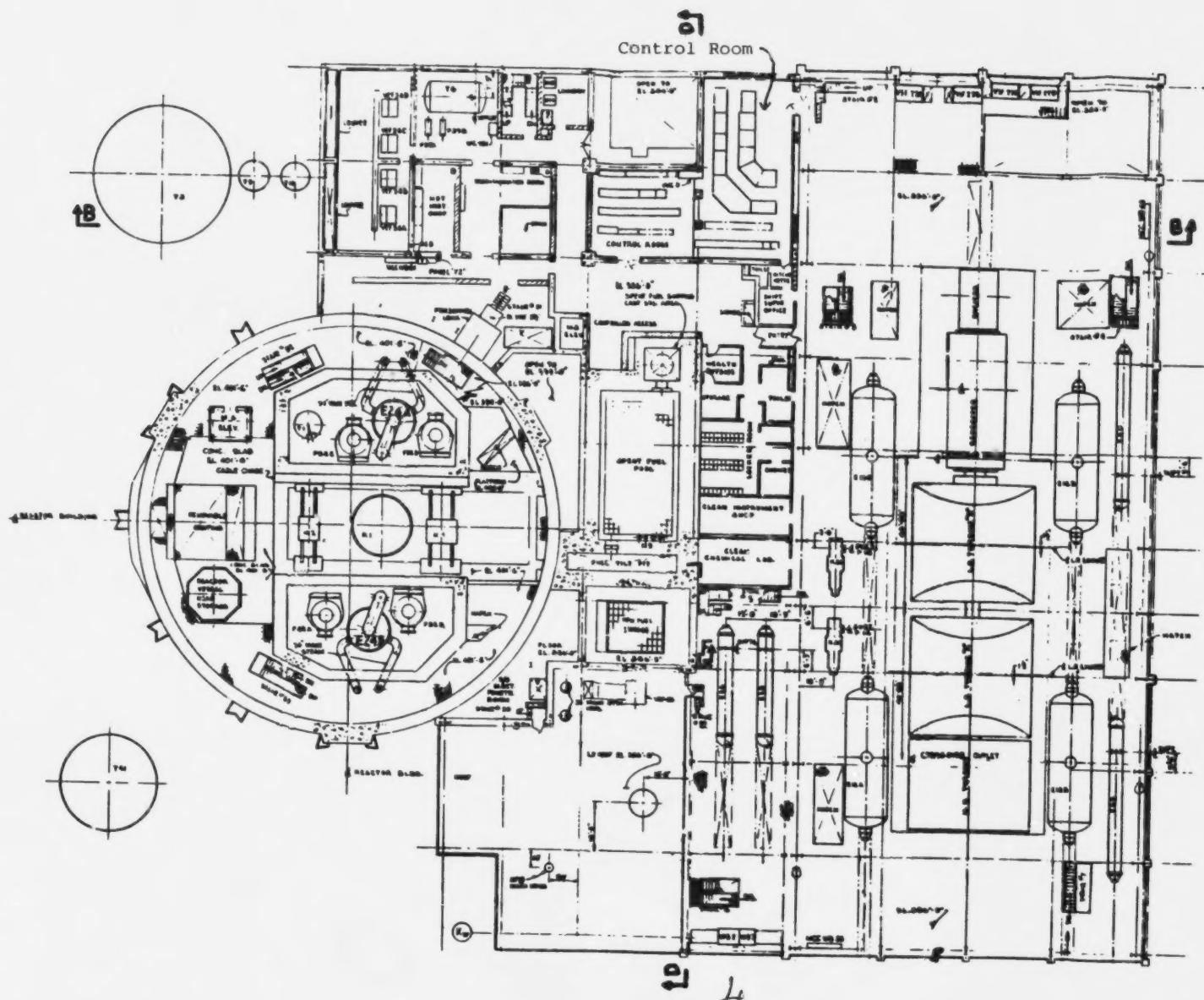
1.4 The Shutdown Decay Heat Removal Analysis

As noted above, the ANO-1 Plant was identified in the initial qualitative screening as having sufficient potential vulnerabilities to warrant additional study. This initial screening and then subsequent identification and evaluation of alternatives for improving DHR are the subjects of this report. Some potential accidents which could result in core melt have not been analyzed. Since the purpose of this program is to study the adequacy of shutdown decay heat removal systems, large LOCAs, reactor vessel ruptures, interfacing system LOCAs, and anticipated transients without scram (ATWS) are excluded. Furthermore, special issues being studied in depth elsewhere, such as pressurized thermal shock, are also excluded.

The flow of the analysis and value-impact assessment are illustrated in Figure 1.7. The internal analysis proceeds along the well-documented approach used for other probabilistic risk assessments (PRA). The potential accident initiating events are identified and combined with required functions in event trees. Then using Boolean solutions to the fault trees describing the safety systems required to perform the safety functions combined with the initiating events, accident sequences are defined which can lead to situations resulting either in safe conditions or core melt. The probability of those sequences occurring which result in core melt was quantified using available computational tools. This quantification also identifies the specific random failure events (e.g., equipment failure, operator error) which contribute to the core melt and the magnitude of that contribution. In addition, for core melt sequences the performance of containment systems was also examined. Then using the results of past PRAs, these results are used to determine the containment failure modes. This leads to delineation of potential release category probabilities which can be used to establish public risk in terms of expected population radiation dose and other measures. The results of the internal analysis are estimates of the probability of specific failures and their contribution to core melt and public risk. This information can then be used to suggest modifications to equipment or procedures which reduce or eliminate those failures. This process is discussed further in Section 2 and Appendix B.

(Text continued on page 1-11.)

Figure 1.1 Operating Floor Level (385')



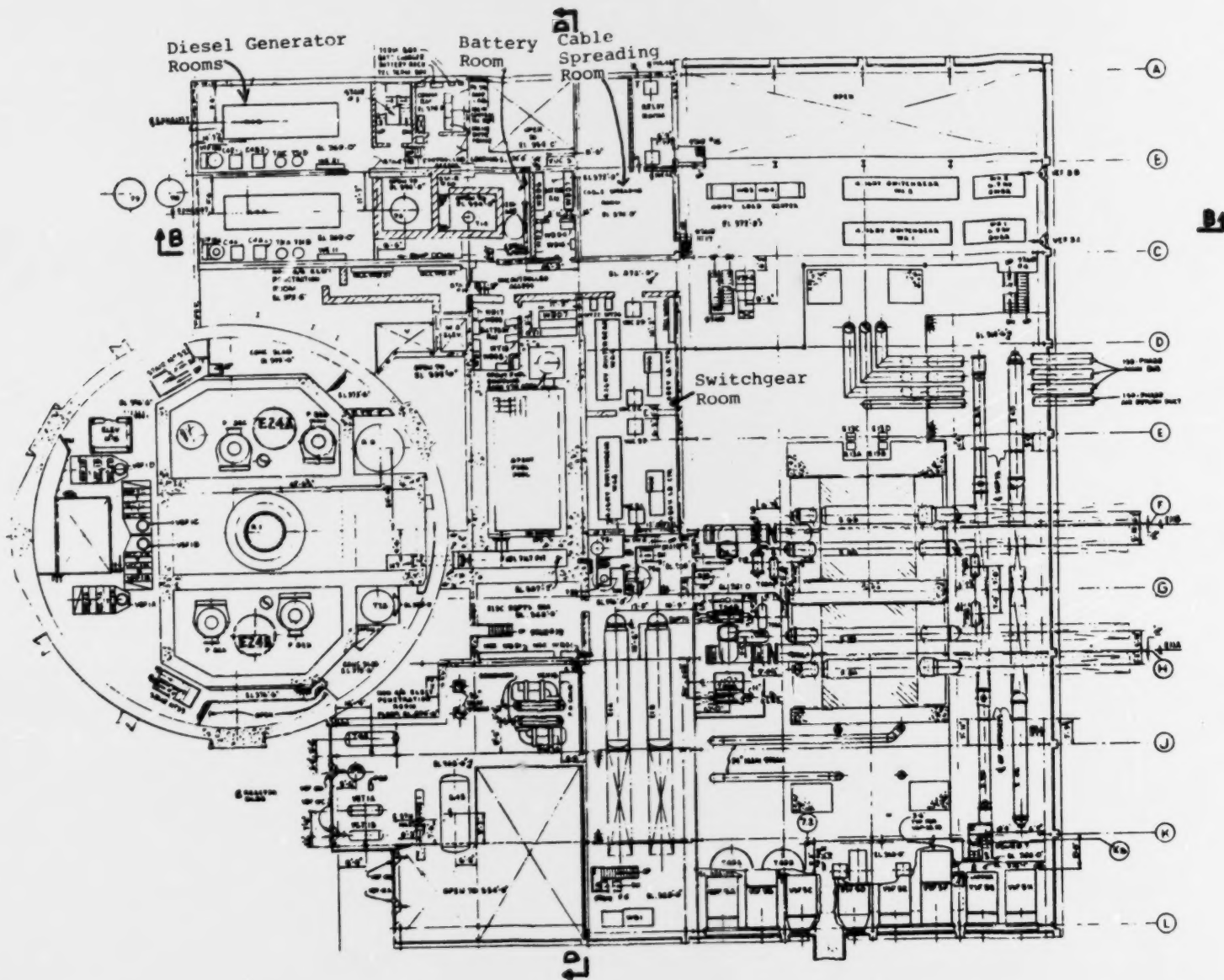


Figure 1.2 Intermediate Floor Level (372')

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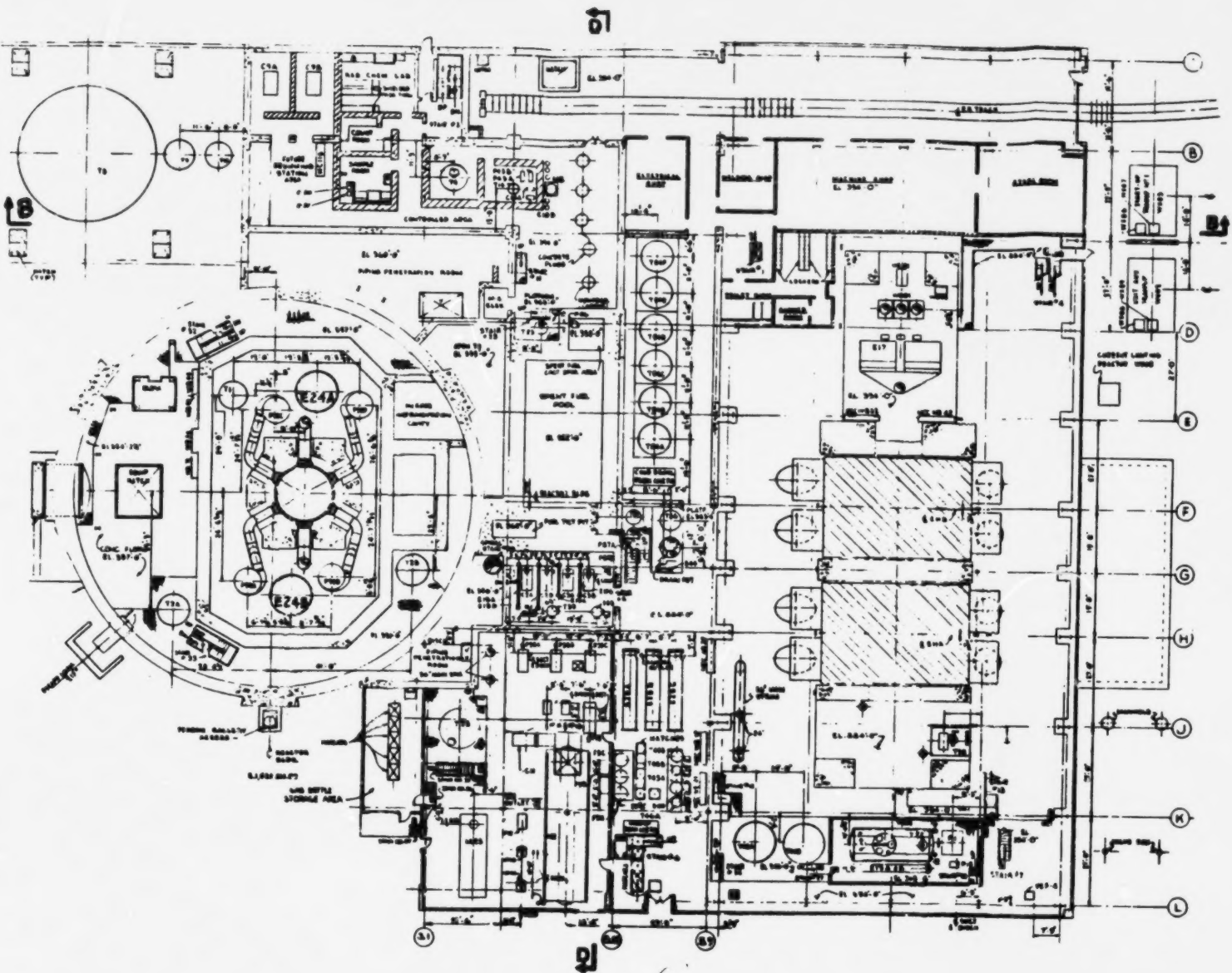


Figure 1.3 Ground Floor Level (354')

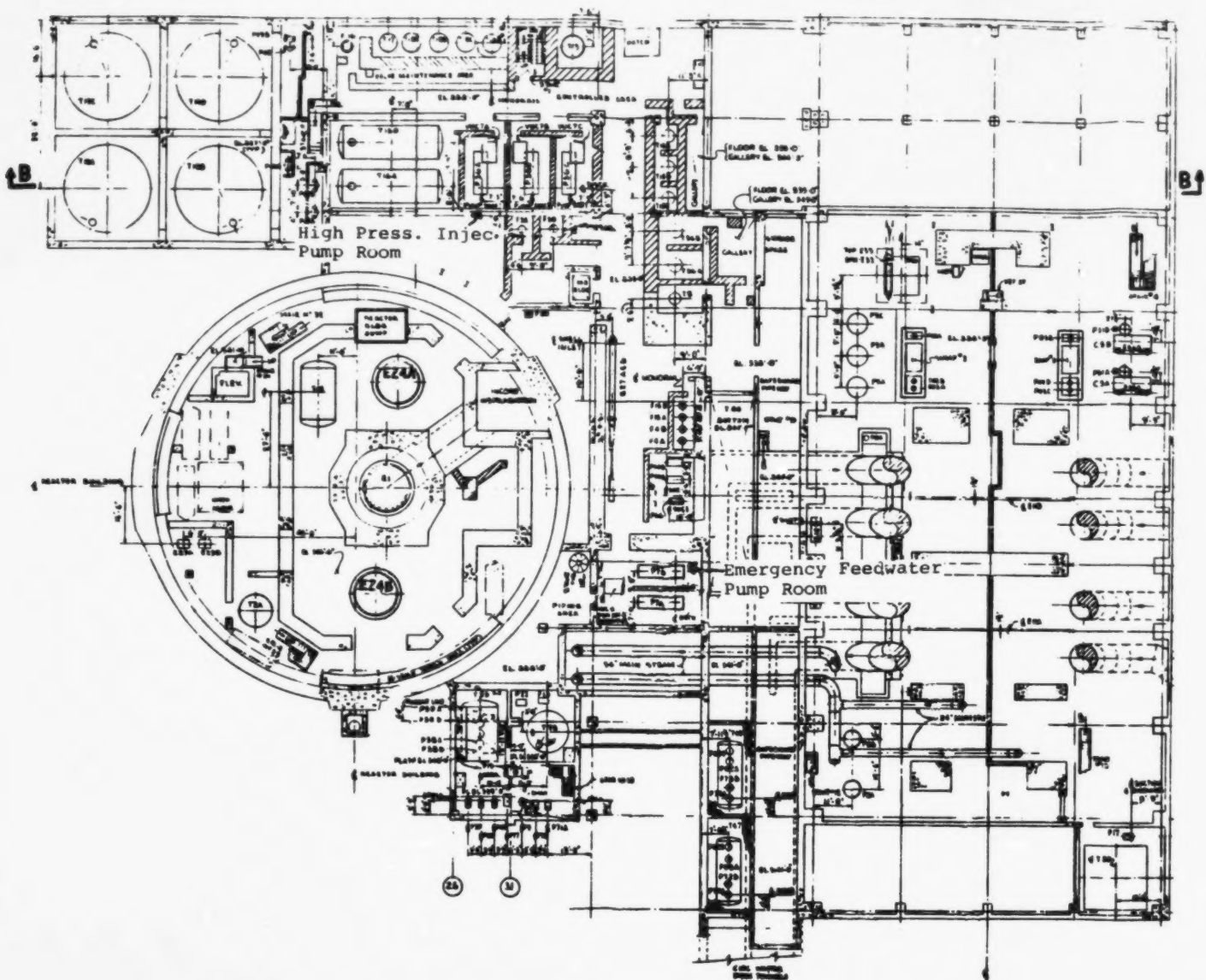
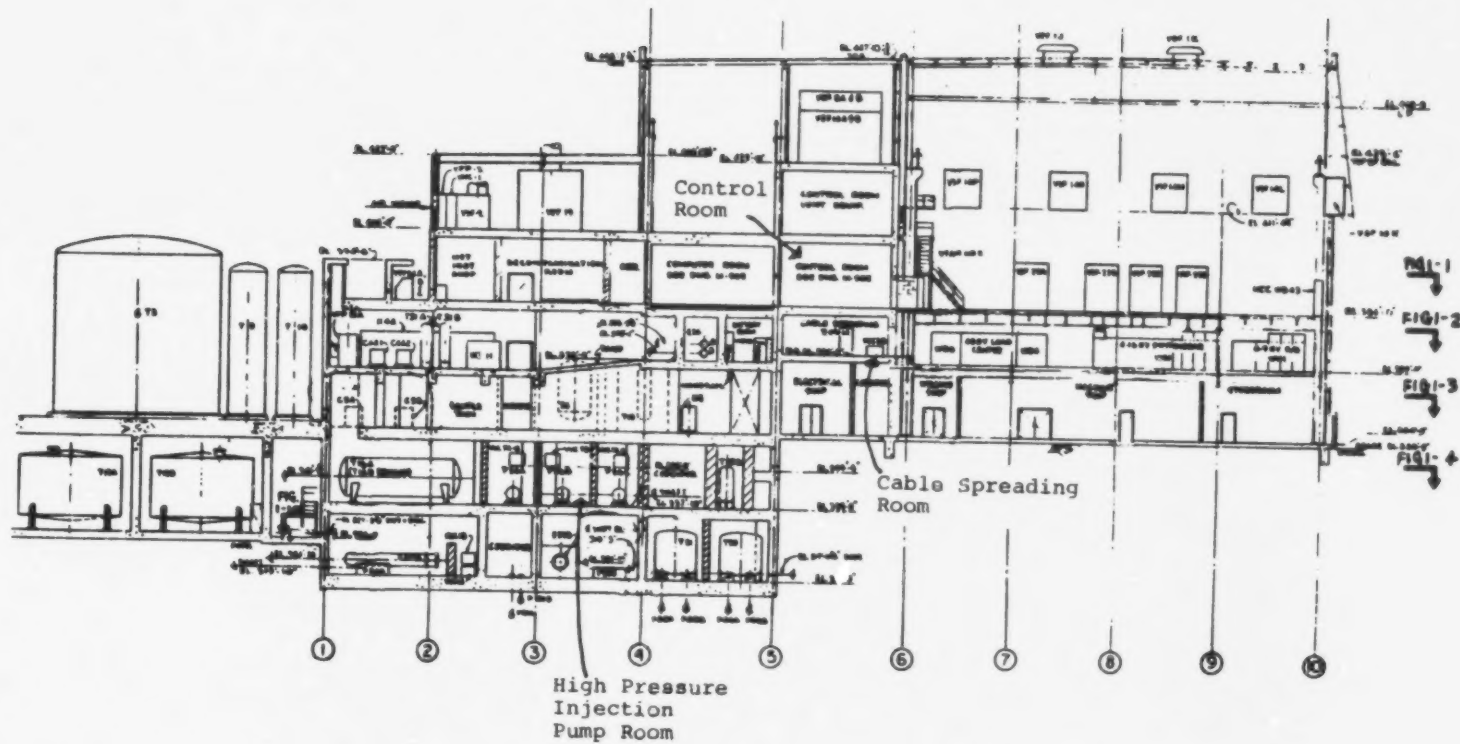


Figure 1.4 Below Grade Level (335')

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Figure 1.5 Cross-section B-B



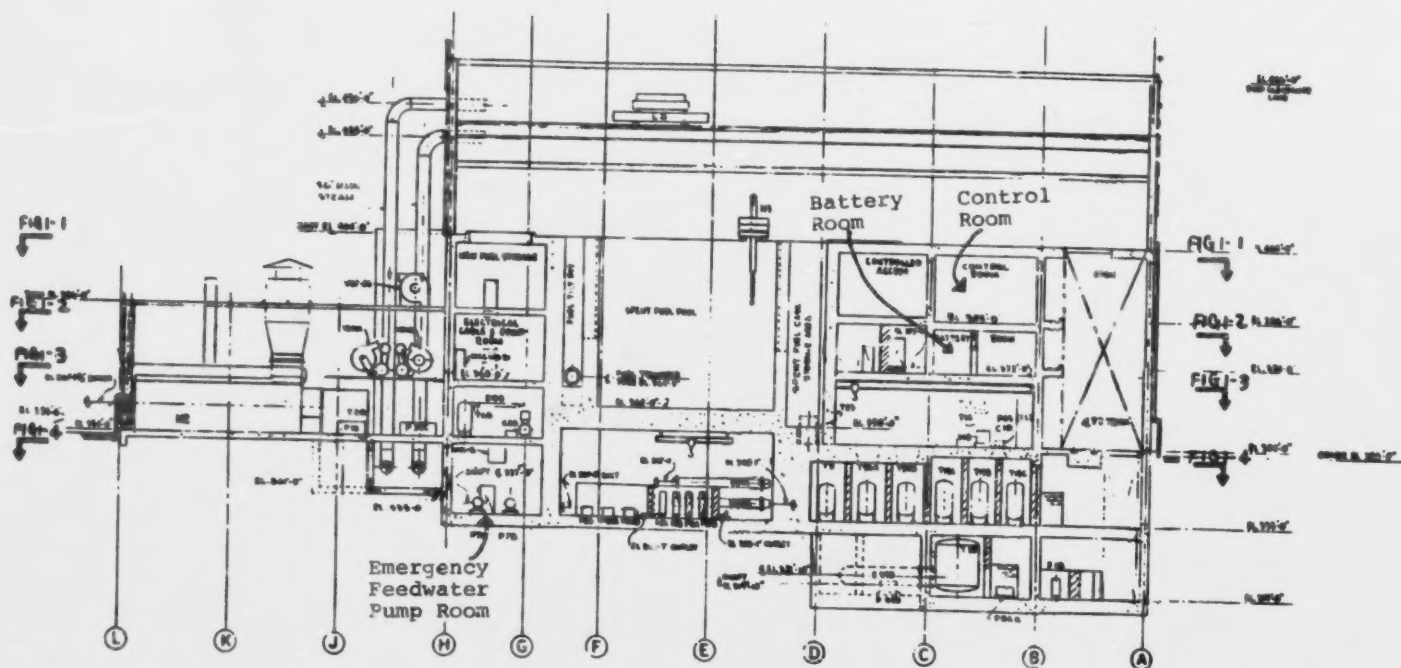


Figure 1.6 Cross-section D-D

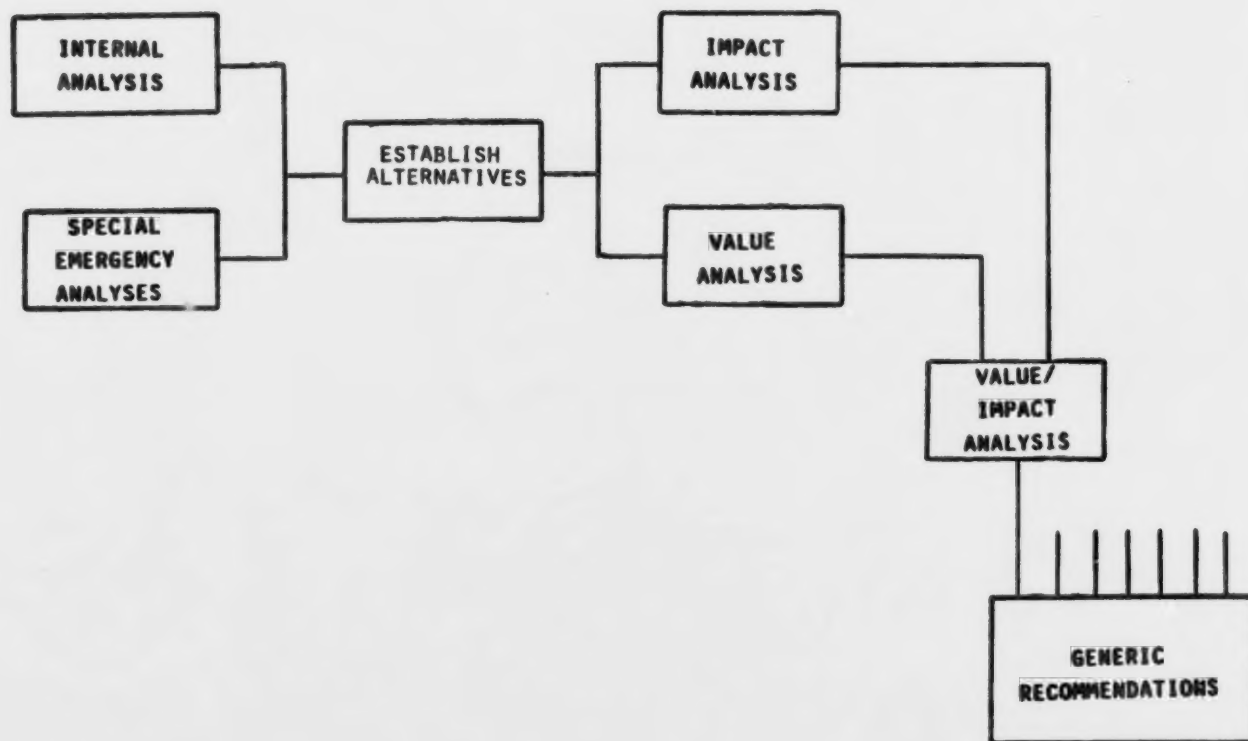


Figure 1.7 Shutdown Decay Heat Removal Analysis

The special emergency analyses proceed on a parallel and similar path although perhaps more qualitative than the internal analysis and certainly more dependent upon engineering judgment. Generally the special analysis for earthquake, flood, and high winds proceed by identifying the hazard and its frequency of occurrence. Then an assessment of the response of the plant to this hazard was established. A key aspect of this assessment is an onsite inspection of the plant and its equipment. Once the response is established, individual equipment fragilities are established. Then using the internal event fault trees and events, the potential contribution of specific failures to core melt probability were estimated. Using this information modifications were defined again to eliminate or reduce the effects of certain vulnerabilities.

The special emergency analyses for fire and internal flood proceed in a slightly different fashion. First, potentially significant fire areas are identified using the transient event trees and system fault trees to establish the critical front line and support systems. Second, the physical arrangements of equipment and potential fire sources were verified by a site visit. The effects of the special emergency were then quantified based on historical fire occurrence frequencies, analytical models of fire growth, and fire suppression probabilities. In addition, random failures and human factors were considered as applicable. Based upon these results, modifications designed to reduce core melt can be proposed.

The special emergency analyses are discussed in Section 3 and Appendices C through I.

Based upon initial estimates of the possible reductions in core melt probability that might be achieved by the various modifications, the modifications are combined into groups for engineering evaluation of feasibility and impact. These combinations are called alternatives. The alternatives are reviewed by the Architectural Engineer (AE) and the plant staff for feasibility and reasonableness. Based upon this initial review, the AE proceeds with conceptual design. After an onsite inspection to insure that the proposed designs could be implemented and to gather site specific cost and related data, the AE develops the impact data. This data includes, but is not limited to, capital costs for engineering and installation, radiation exposures incurred, maintenance and operational costs. Once the conceptual designs are reasonably well established, the internal and special emergency analyses are rerun for each alternative to establish their value in terms of reduced core melt probability and reduced public risk. These analyses are described in Sections 6 and 7 respectively. The impacts and values are then integrated into a value-impact analysis. This analysis is structured according to existing NRC guidelines. Further discussion of this analysis is provided in Section 9. A brief uncertainty analysis is

provided in Section 8 and an analysis of special issues (i.e., bleed and feed, and secondary blowdown mode of operation) is provided in Section 10. Finally, the analysis suggests those generic insights or conclusions that might be inferred from the results for one plant (Section 11).

2.0 INTERNAL ANALYSIS - METHODS AND RESULTS

2.1 Analysis Methods

The internal analysis methods are described in this section using the pictorial representation of the internal analysis shown in Figure 2.1 as a guide. After the various tasks are explained an example will be given in the symbols that are typical of probabilistic risk assessments and of this analysis for ANO-1.

2.1.1 Modeling

The process begins with the selection of the initiating events to be considered. Since the purpose of this program is to study the shutdown decay heat removal (SDHR) function adequacy, large LOCAs, reactor vessel ruptures, interfacing system LOCAs, and anticipated transients without scram (ATWS) were excluded. Furthermore, special issues being studied elsewhere such as pressurized thermal shock were also excluded. Thus, the initiating events to be analyzed are:

- Small LOCA
- Loss of Offsite Power Transients
- Transients resulting from initial Loss of the Power Conversion System (PCS)
- Transients with offsite power and the PCS initially available
- Transients resulting from the loss of an AC or DC bus

In order to develop the event trees for each of these initiating events, the success criteria for the systems contributing to the SDHR function were delineated using the plant Final Safety Analysis Report and discussions with plant personnel as the sources of information. The objective was to define the success criteria and potential options to address the various accident conditions realistically and not overly conservative. The front line systems that were considered are:

- Emergency Feedwater System (L)
- High Pressure Injection (D_1) & Recirculation System (H_1)
- Low Pressure Injection (D_2) & Recirculation System (H_2)
- Power Conversion System (M)
- Pressurizer Safety and Relief Valves (P)
- Secondary Safety and Relief Valves

In addition, the support systems considered are:

- Service Water System
- Emergency Safeguards Actuation System
- Electric Power System

Each of the front line systems or combinations of those systems, e.g., feed and bleed mode (E) using the high pressure system, manually initiated, becomes a column heading or decision branch in the event trees. The event tree starts with an initiating event and successively asks whether or not the systems (in each of the headings) are successful or fail. In some branches previous success/failure answers make later decisions irrelevant resulting in an asymmetric event tree that most simply represents the sequence of events leading to the potential outcomes, success or core melt. A transient event tree is shown in Figure 2.1 in the upper left hand part of the diagram. A LOCA event tree is shown to the right of the transient event tree representation. LOCAs may be initiated independently or can result from transient induced LOCAs as depicted in the diagram. The final result in any of the cases, transient, transient induced LOCA, or LOCA, is an accident sequence made up of the success or failure of events (i.e., event tree headings representing front line systems) leading to either success or core melt.

The accident sequences will be carried further into the analysis, but first it is important to look at the development of the event tree headings. These are modeled on fault trees such as the one shown in the lower left side of Figure 2.1. Each fault tree is composed of all the components in the flow paths of the systems involved in an event tree heading. Often the event tree heading is simply one specific system. Sometimes more than one system may be combined in a fault tree. In any case the fault tree represents all the potential failure combinations that could lead to failure of the system based on its system success criteria, e.g., flow from 1 out of 3 pump trains. The component is the key to the development of the fault tree. At the component level all the ways that component can fail are modeled including the support systems required by that component. Interactions between components and trains of a system are implicitly modeled by the step-by-step process of constructing the fault tree and identifying all the potential failures.

In this program the event trees are small and the fault tree medium in size which reflects the detail permitted by the scope of the program. This seemed to be the proper mix to meet the objectives of the program.

The outcome of each branch of an event tree is an accident sequence such as $S_2MLD_1P_1$, where the initiating event is a small LOCA followed by loss of the PCS, failure of the auxiliary feedwater system, success of the high pressure injection system, and finally, failure of the pressurizer power operated relief valves leading to an early core melt.

2-3

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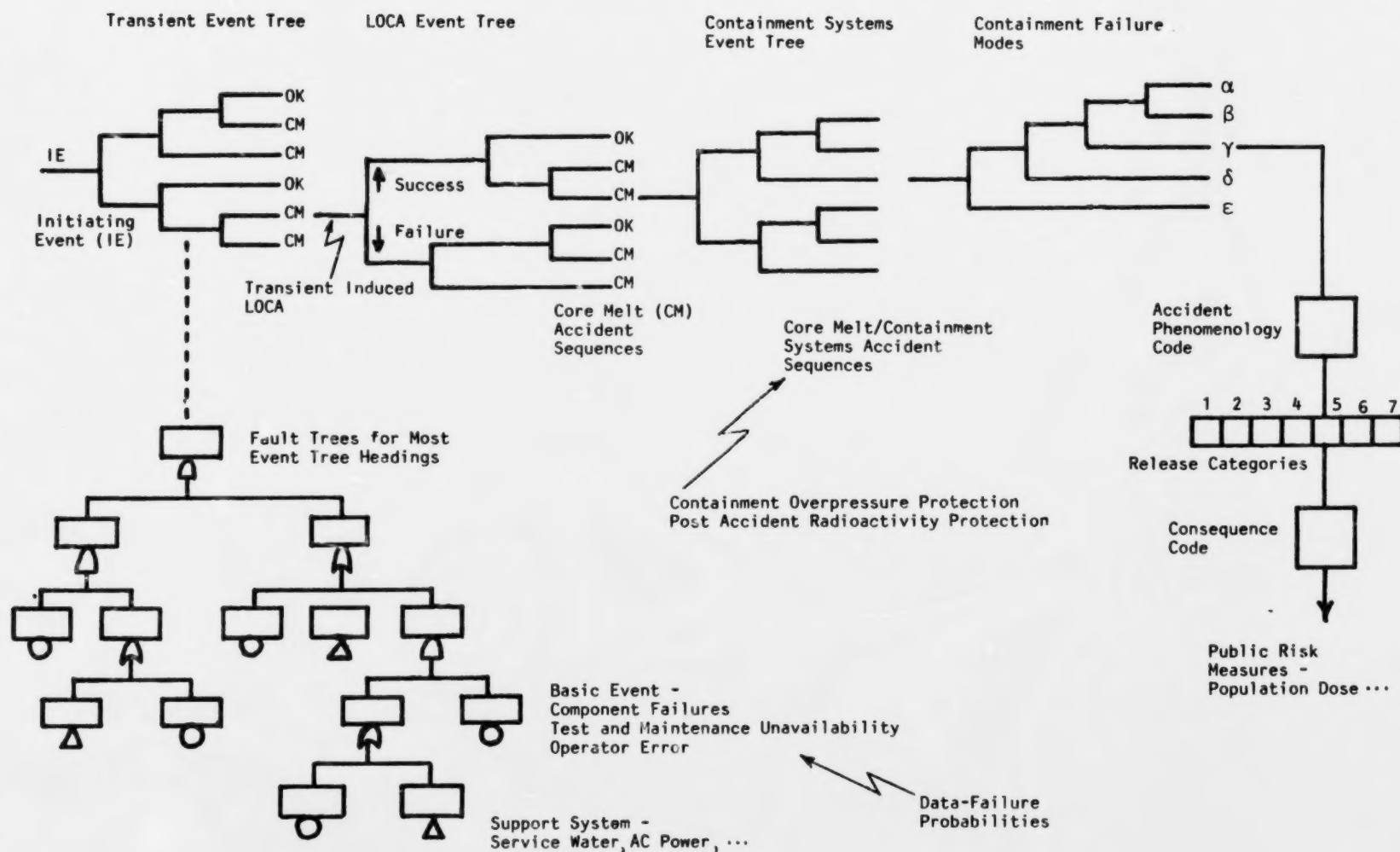


Figure 2.1 Pictorial Representation of the Internal Analysis

The last part of the modeling task is to establish the data base to be used to quantify the accident sequences. Each initiating event frequency and basic fault tree event probability must be derived from data or engineering judgment. The data used in this analysis is essentially generic in nature. Within the analysis some failure probabilities apply to many components. For example, the local fault (LF) of a motor operated valve (V) which is normally closed (NC) and fails to open, i.e., fails closed (FC), VCC-LF is $8E-3$ per demand. Thus LP1407A-VCC-LF, EF000Y1A-VCC-LF, SWS3806A-VCC-LF, etc., would all be assigned the same failure probability.

The model then consists of event trees for each initiating event, fault trees for each event used in the event trees, and the data base (or value block in the computer program terminology).

2.1.2 Accident Sequence Analysis

The output of the SDHR model is a set of accident sequences in which each accident sequence is composed of the product of the Boolean equation for each of the accident sequence events. In the ANO-1 analysis there were 60 distinct accident sequences identified potentially leading to core melt. The SETS Boolean reduction computer code (reference 6) was run on each of these accident sequences to obtain the core melt probability and the significant combinations of basic events that result in core melt. The Boolean reduction results in a sum of products of basic events. Each of these products is called a cut set such as PUMP1 times VALVE7 times OPERATOR ACTION A.

Every accident sequence is made up of unmodeled and modeled events. For example, in the accident sequence $S_2ML\bar{D}_1P_1$, the S_2M events are not modeled but are assigned a frequency of S_2 and a probability for M from the data base. The events $\bar{D}_1P_1$ are the Boolean part of the accident sequence leading to the cut sets. The product of $f(S_2)*p(M)*p(\bar{D}_1P_1)$ is the core melt contribution for this accident sequence. The sum of the core melt probabilities for all 60 accident sequences is the SDHR core melt probability. However, there are recovery actions that can be taken which were not included in the basic model due to the complexity that would have been incurred. Therefore, the next step is to apply the appropriate recovery factors to all the significant cut sets of the dominant accident sequences (that is the accident sequences that contribute the most to core melt ~ 90 + %). There were 23 accident sequences in the ANO-1 analysis carried through the recovery analysis. Application of recovery results in a lower SDHR core melt probability.

2.1.3 Vulnerability Identification

The most significant cut sets of the dominant accident sequences after recovery has been applied are the basis for identifying

the potential internal vulnerabilities. When these cut sets are examined and grouped by types of failures a pattern develops in that certain basic event failures appear in the cut sets from several accident sequences. The outcome is a ranking of potential vulnerabilities, the cause of each vulnerability and the contribution of each vulnerability to the total SDHR core melt probability. The objective is to identify those vulnerabilities which contribute something over 80% of the total SDHR core melt probability and then address possible modifications to reduce or eliminate them.

A further screening of the accident sequences is provided by the vulnerability identification resulting in only 14 accident sequences requiring further consideration. While the vulnerabilities were selected based on core melt probability, the input needed for the value-impact analysis is risk to the public. The risk measure chosen is population dose in person-rem. In order to obtain population dose the dominant accident sequences must be extended to include containment failure, environmental transport, and consequence analyses.

2.1.4 Containment System Integration

In past PRAs it was assumed that failure of the containment resulted in sump flashing and subsequent failure of the emergency core cooling systems due to pump cavitation. Based upon discussions with several analysts and utility personnel, this is an overly conservative approach, thus in this analysis core melt does not depend on containment systems. The containment systems event tree for ANO-1 is depicted in Figure 2.1 on the extension of one of the core melt (CM) accident sequences. Each of the 14 dominant accident sequences was rerun with each of the 6 Boolean containment systems event tree outcomes to establish the probabilities for the containment systems outcomes or states. This is the input for the next stage of the analysis.

The containment systems considered are:

Containment Spray Injection (C) and Recirculation (F) System
Reactor Building Cooling System (Y)

Outcomes of the containment systems event tree are success or failure of the containment overpressure protection function and success or failure of the post accident radioactivity removal function. These outcomes and the early/late core melt state determine the release category assignment for each accident sequence.

After each of the 14 dominant accident sequences were run with each of the 6 containment systems sequences (3 for transients), a further screening was performed resulting in only 38 sequences (vs 69) for ANO-1 as candidates for the next recovery

analysis. Another recovery analysis is needed since previous recovery analysis could not be conveniently brought forward nor would it necessarily be applicable since the cut sets change in the new Boolean reduction.

2.1.5 Interface with Accident Phenomenology

The 38 sequences to be assigned to the standard release categories each consists of an initiating event, a core melt event sequence, and a containment systems event sequence. The result of past PRA accident phenomenological codes were reviewed for applicability to ANO-1 and a table developed to map each containment system/early-late core melt state into a release category. This mapping assigns a containment failure mode and apportions the containment failure between these modes (the sum plus the probability of no containment failure equals 1). The containment failure modes are:

α	In-vessel steam explosion
β	Containment leakage
γ	Hydrogen burn overpressure
δ_e	Ex-vessel steam spike
δ_q	Steam and noncondensable gas overpressure
ϵ	Base mat melt through

The sum of the probabilities for the sequences placed in each release category is then combined with the corresponding results from the special emergencies. Finally, the CRAC consequence code (reference 35) was run for the ANO-1 site providing a population dose factor (out to 50 miles radius) for each release category. Thus the total population dose (out to 50 miles radius) equals the sum of the probability of release category i * population dose i for $i = 1$ to 7. This is used in the value-impact analysis.

2.1.6 Value Analysis

The core melt probability and population dose described in the previous subsections comprise the base case. The modifications proposed for each of the vulnerabilities identified can be grouped along with special emergency modifications to form alternatives. The value of an alternative is the decrease in core melt probability or the decrease in population dose from the base case. Therefore the entire process is repeated for each alternative. Fortunately, there are some shortcuts but essentially it is the same method. In the ANO-1 analysis there were only 2 alternatives selected.

2.1.7 Example

It is instructive to follow one sequence from the initiating event all the way through to the population dose and referencing the tables or figures in the analysis where each step occurs. This accident sequence is:

$$T_2MQ\bar{L}\bar{D}_1H_1H_2' - \bar{Z}_L\bar{C}F-\alpha$$

This is a transient initiating event (T_2) caused by loss of the power conversion system (M), with success of the auxiliary feedwater system ($\bar{L}$), but failure of the pressurizer safety valves to reclose after opening although not demanded, which leads to a transient induced LOCA (see Figure B.2 branch 1). The sequence then enters the LOCA event tree where the auxiliary feedwater system is successful ($\bar{L}$). The high pressure injection system is successful ($\bar{D}_1$) but both the high pressure recirculation system (H_1) and the low pressure recirculation system (H_2') fail leading to a late core melt. The prime on the H_2 event indicates that the LPRS heat exchangers are not required according to the success criteria, since the APWS is successful.

Given core melt, the containment system response is success of the containment overpressure protection function ($\bar{Z}_L$), success of the containment spray injection system ($\bar{C}$), but failure of the containment spray recirculation system (F). Note that the containment spray recirculation system depends on the low pressure recirculation system for suction pressure from the sump. The containment event tree is given in Figure B.5 where Z_L is composed of the containment fans and sprays.

The core melt probability is

$$p(T_2MQ) * p(\bar{L}\bar{D}_1H_1H_2') = (1.4E-3) * (1.2E-3) = 1.7E-6$$

where the multiplier frequency is found in Table B.1 and the Boolean accident sequence probability is found in Table B.11 sequence number 27 or in Table B.12 which is a summary table. The probability of the sequence including the containment systems is

$$p(T_2MQ\bar{L}\bar{D}_1H_1H_2' - \bar{Z}_L\bar{C}F) = 5.7E-7$$

found in Table B.27.

This is the end of the internal analysis but it is useful to carry the sequence through to its population dose. The sequence state is a late core melt, containment overpressure success ($\bar{Z}_L$) and a late containment spray failure ($\bar{C}F$). Table 7.2 indicates that the α containment failure mode maps into release category 1 with a probability of $1E-4$. Thus the complete sequence probability is

$$\begin{aligned} p(T_2MQH_1H_2' - \bar{Z}_L\bar{C}F-\alpha) &= p(TMQH_1H_2' - \bar{Z}_L\bar{C}F) * P(\alpha) \\ &= (5.7E-7) * (1E-4) = 5.7E-11 \end{aligned}$$

with the success events not shown as is customary. The corresponding population dose for the baseline source term case (Table 7.5) is,

$$(3.6E-11) * (4.7E+5) = 1.7E-5 \text{ person rem.}$$

2.2 Core Melt Probability Results

The ANO-1 model started with 60 accident sequences and through several screening steps reduced to 14 accident sequences that were important. These results are summarized in Table 2.1. The details of the analysis are found in Appendix B. The vulnerabilities identified are described in Section 2.3.

These 14 sequences can be grouped according to the type failures leading to core melt.

- a) $S_3^{MH_1H_2'}$ - small LOCA and failure of ECC recirculation
 $p = 2.2E-5$
- b) T_1^{MLE}
 T_2^{MLE}
 T_3^{MLE}
 T_4^{MLE}
 T_5^{MLE} - Transient with loss of emergency feedwater and failure of feed and bleed
 $p = 3.1E-5$
- c) $T_3^{MQH_1H_2'}$
 $T_2^{MQH_1H_2'}$
 $T_3^{QH_1H_2'}$ - Transient induced LOCA and failure of ECC recirculation
 $p = 1.5E-5$
- d) $T_2^{MQD_1D_2}$
 $T_3^{QD_1D_2}$ - Transient induced LOCA and failure of ECC injection
 $p = 5.2E-6$
- e) $S_3^{MED_2}$ - small LOCA and failure of ECC injection
 $p = 6.6E-6$
- f) $T_1^{MQLD_1}$ - Transient induced LOCA with loss of emergency feedwater and failure of ECC injection
 $p = 7.8E-7$
- g) S_3^{MXE} - Small LOCA with failure of ECC injection and failure to achieve secondary blowdown
 $p = 2.3E-6$

Table 2.1

ANO-1 Core Melt Accident Sequences

Analysis Number	Dominant Accident Sequences	Probability Before Recovery	Probability After Recovery
8	$S_3MH_1H_2'$	1.15E-4	2.24E-5
15	T_1MLE	7.27E-5	2.13E-5
39	$T_3QH_1H_2'$	5.71E-5	1.21E-5
10	S_3MED_2	5.43E-5	6.60E-6
41	$T_3QD_1D_2$	2.74E-5	4.57E-6
51	T_4MLE	1.32E-5	4.08E-6
59	T_5MLE	9.64E-6	2.46E-6
17	T_2MLE	5.66E-5	2.38E-6
11	S_3MXE	4.55E-6	2.33E-6
27	$T_2MQH_1H_2'$	8.07E-6	1.71E-6
19	T_3MLE	2.42E-5	1.13E-6
26	T_1MQLD_1	1.58E-6	7.84E-7
33	$T_3MQH_1H_2'$	3.46E-6	7.32E-7
29	$T_2MQD_1D_2$	3.99E-6	6.66E-7
Top 14 accident sequences based on probability after recovery applied		4.521E-4	8.319E-5
All accident sequences		4.720E-4	8.758E-5

Approximate overall recovery factor of 5.4.

The most important sequences are transients with loss of emergency feedwater and failure of feed and bleed, followed by small LOCAs and failure of ECR, and transient induced LOCAs with ECR Failure. Analyses described in Section 2.3 will show that valve common mode, battery common mode, and failure of the low pressure injection pumps are the dominant cut sets for these sequences.

2.3 Potential Internal Vulnerabilities and Modifications

Eight potential vulnerabilities were identified from the internal analysis for further consideration. Briefly, the procedure was to first examine the cut sets of the dominant accident sequences generated from the fault tree models and generic failure data. A recovery analysis was then performed on all the significant cut sets to give credit for normal operator actions and emergency procedures. The significant cut sets remaining constitute the potential vulnerabilities described here for further examination and possible suggested modifications. This procedure provides a screening process to focus attention on the most significant cut sets and specific basic events in those cut sets.

The following descriptions first identify the cut set which may be one or more basic events and the accident sequences to which that cut set contributes. Each accident sequence is made up of events that were not modeled, called the multiplier, and a Boolean product of events that were modeled. The Boolean expression is made up of numerous cut sets. The probability of each of these cut sets times the probability of the multiplier is a contribution to the internal analysis core melt probability. The cut sets to be discussed below are those cut sets which contribute significantly to the overall core melt probability in one or more accident sequences.

A cut set may appear in more than one accident sequence. A cut set probability is usually the same each time it appears (the exception being where it is dependent on an event in the multiplier expression) but its contribution to the overall core melt probability is the cut set probability times the particular accident sequence multiplier probability. This contribution is given in the vulnerability descriptions to the right of each dominant accident sequence in which that cut set appears. The sum of the contributions from each dominant accident sequence in which that cut set appears represents the total effect of that cut set on core melt probability for the plant based upon the internal analysis. A brief description of the accident sequence events is given in Table 2.2 as a reference.

The description explains the basic events in the cut set and why it is important to core melt. The suggested modification is the instruction to the architectural engineer for an impact

analysis. This may have been modified during subsequent discussions regarding feasibility of the modification and validity of the problem identified.

INTERNAL VULNERABILITY 1: Failure of the Emergency Feedwater system turbine driven Pump along with other safety system failures.

APPLICABLE CUT SETS:

EFWOP7AX-PTD-LF * various other single failures

Dominant accident sequences affected and estimated core melt probability contribution resulting from this vulnerability/cut sets

T ₄ MLE	3.40E-6
T ₅ MLE	1.88E-6
T ₂ MLE	8.71E-7
T ₃ MLE	4.15E-7
T ₁ MLE	<u>1.93E-7</u>
	6.77E-6

DESCRIPTION: This vulnerability is the combination of the failure of the Emergency Feedwater turbine driven pump due to local faults with the single failure of various other components. The turbine pump is the recurring failure that establishes this vulnerability.

MODIFICATION: Use the Auxiliary Feedwater motor driven pump as a backup to the EFWS. To do this the AFW pump must be connected to class 1E power and a direct line to the CST must be installed (see Figure 2.2). If necessary, the pump should be modified for self-cooling of the lube oil. This modification also addresses vulnerability 4.

Table 2.2
Accident Sequence Events

Initiating Events

S ₂	Small LOCA 1.2" to 1.66" Diameter
S ₃	Small LOCA 0.38" to 1.2" Diameter
T ₁	Loss of Offsite Power Transient
T ₂	Loss of Power Conversion System
T ₃	Transients with Main Feedwater Initially Available
T ₄	Transient initiated by loss of AC Bus
T ₅	Transient initiated by loss of DC Bus

System Events

M	Failure of Main Feedwater
Q	Failure of SRVs to Reclose Given That They Were Opened
L	Failure of the Emergency Feedwater System
E	Failure of Feed and Bleed Mode of HPIS
D ₁	Failure of HPIS
D ₂	Failure of LPIS
H ₁	Failure of HPRS With RHR Heat Exchangers
H ₂ '	Failure of LPRS Without RHR Heat Exchangers

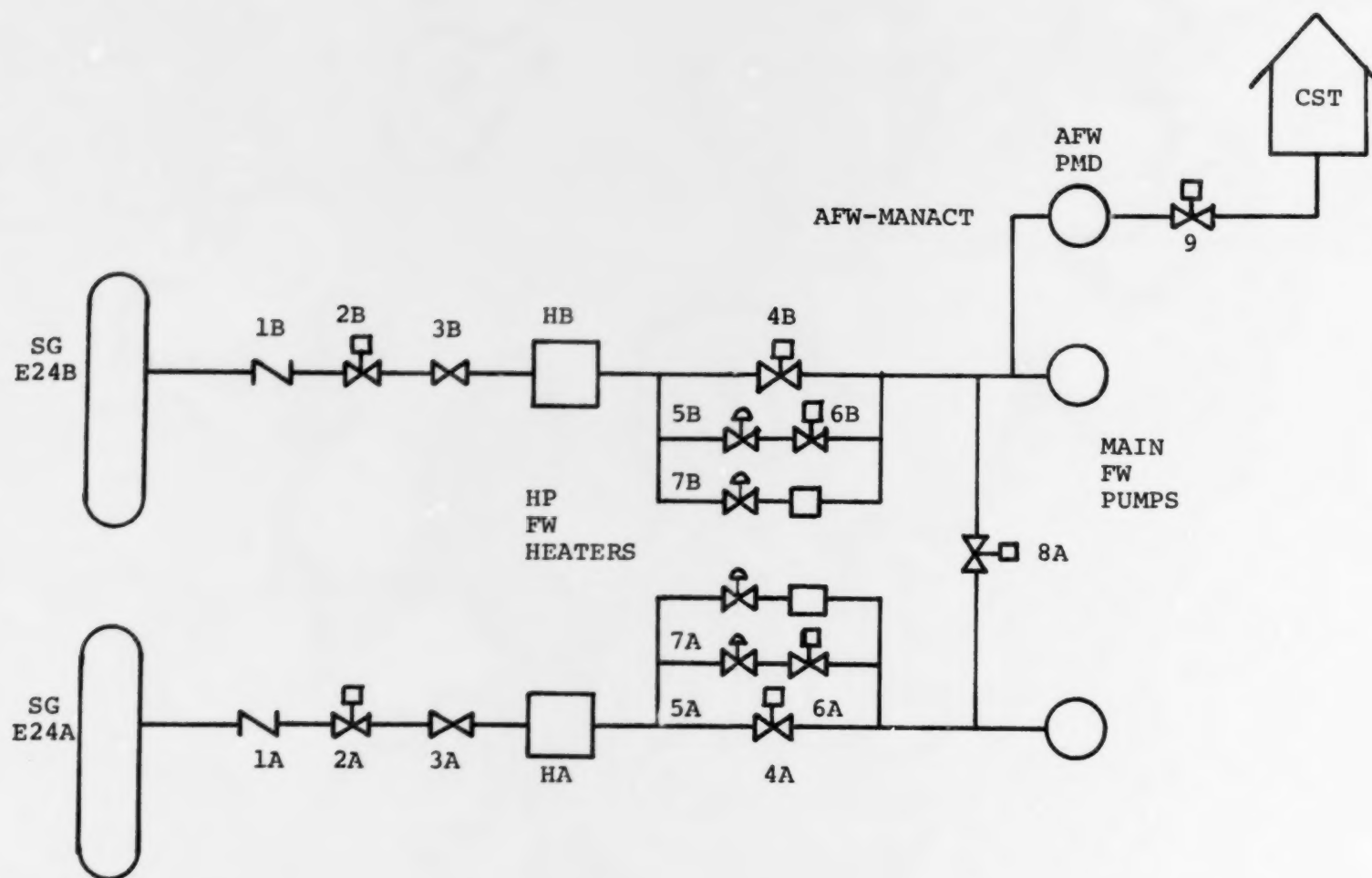


Figure 2.2 Modification for Vulnerabilities 1 and 4, ANO-1

INTERNAL VULNERABILITY 2: Common mode failure of valves in various safety systems.

APPLICABLE CUT SETS:

HPISV-CM	5.42E-6
LRSV-CM	4.85E-6
SWSV-40-41-CM	4.85E-6
SWSV-02-03-CM	<u>4.85E-6</u>
	2.01E-5

Dominant accident sequences affected and estimated core melt probability contribution resulting from this vulnerability/cut sets

S ₃ MH ₁ H ₂ '	9.12E-6
T ₃ QH ₁ H ₂ '	4.52E-6
S ₃ MED ₂	2.96E-6
T ₃ QD ₁ D ₂	2.15E-6
T ₂ MQH ₁ H ₂ '	6.39E-7
T ₂ MQD ₁ D ₂	3.14E-7
T ₃ MQH ₁ H ₂ '	2.73E-7
S ₃ MXE	<u>1.29E-7</u>
	2.01E-5

DESCRIPTION: The common mode failure of valves are as follows:

SWSV-02-03-CM low pressure pump room coolers
SWSV-40-41-CM low pressure pump lube oil coolers
HPISV-CM BWST discharge valves to ECC system trains
LPRSV-CM ECC suction valves to the RB sump

Each of the valve contributions fail one or more safety systems.

MODIFICATION: Due to the uncertainty about the fundamental cause of valve common mode failure, no modifications are proposed.

INTERNAL VULNERABILITY 3: Common mode failure of safety system pumps.

APPLICABLE CUT SETS:

LPSP-CM	2.23E-6
SWSP-CM	3.00E-7
HPSP-CM	<u>1.98E-8</u>
	2.56E-6

Dominant accident sequences affected and estimated core melt probability contribution resulting from this vulnerability/cut sets

S ₃ MH ₁ H ₂ '	1.40E-6
T ₃ QH ₁ H ₂	6.93E-7
S ₃ MED ₂	1.90E-7
Other	<u>2.80E-7</u>
	2.56E-6

DESCRIPTION: The common mode failure of low pressure, service water, or high pressure pumps can lead to core melt given the initiating events shown above.

MODIFICATIONS: No modifications are proposed for this vulnerability due to its low probability and the unknown factors causing the common mode failure of pumps.

INTERNAL VULNERABILITY 4: Diesel generator local fault and common mode failures with failure of the Emergency Feedwater turbine driven pump.

APPLICABLE CUT SETS:

DG-CM*EFWOP7AX-PTD-LF	1.82E-6
DG1-GEN-LF*DG2-GEN-LF*EFWOP7AX-PTD-LF	1.76E-6
DG1-GEN-LF or DG-2-GEN-LF*EFWOP7AX-PTD-LF*	
various other single failures	<u>1.67E-6</u>
	5.25E-6

Dominant accident sequences affected and estimated core melt probability contribution resulting from this vulnerability/cut sets

T ₁ MI.E	5.05E-6
T ₁ MQLD ₁	<u>1.98E-7</u>
	5.25E-6

DESCRIPTION: Failure of emergency power given a loss of offsite power with failure of the emergency feedwater turbine driven pump leading to core melt.

MODIFICATION: The modification proposed under Vulnerability 1 would also be applicable.

This vulnerability could have been combined with vulnerability 1, however, it was thought to be useful to show the important subset where the diesel generators are involved. If the modification for vulnerability 1 had not been feasible, then a modification addressing the diesel generators could have been explored.

INTERNAL VULNERABILITY 5: Battery local fault and common mode failures.

APPLICABLE CUT SETS:

BATT-CM	1.53E-5
BATT-CM*LOSS-OSP	2.29E-7
BATTD07-ZBT-LF*BATTD06-ZBT-LF	<u>9.10E-8</u>
	1.56E-5

Dominant accident sequences affected and estimated core melt probability contribution resulting from this vulnerability/cut sets

T ₁ MLE	1.48E-5
T ₁ MQLD ₁	5.24E-7
T ₂ MLE	<u>1.55E-7</u>
	1.56E-5

DESCRIPTION: Loss of both batteries due to common mode results in loss of instrumentation and control including failure to start the diesel generators.

MODIFICATION: The proposed modification for vulnerability 5 is a turbine driven generator to supply vital DC, and instrumentation and control for the turbine driven EFW system pump. In addition this source would supply power to the Auxiliary Feedwater pump proposed in vulnerability 1 above (Figure 2.3). This application of the turbine driven generator addresses the need for AC power for the auxiliary feedwater pump given station blackout. The generator should have a dedicated battery for field flashing or not require additional field flashing.

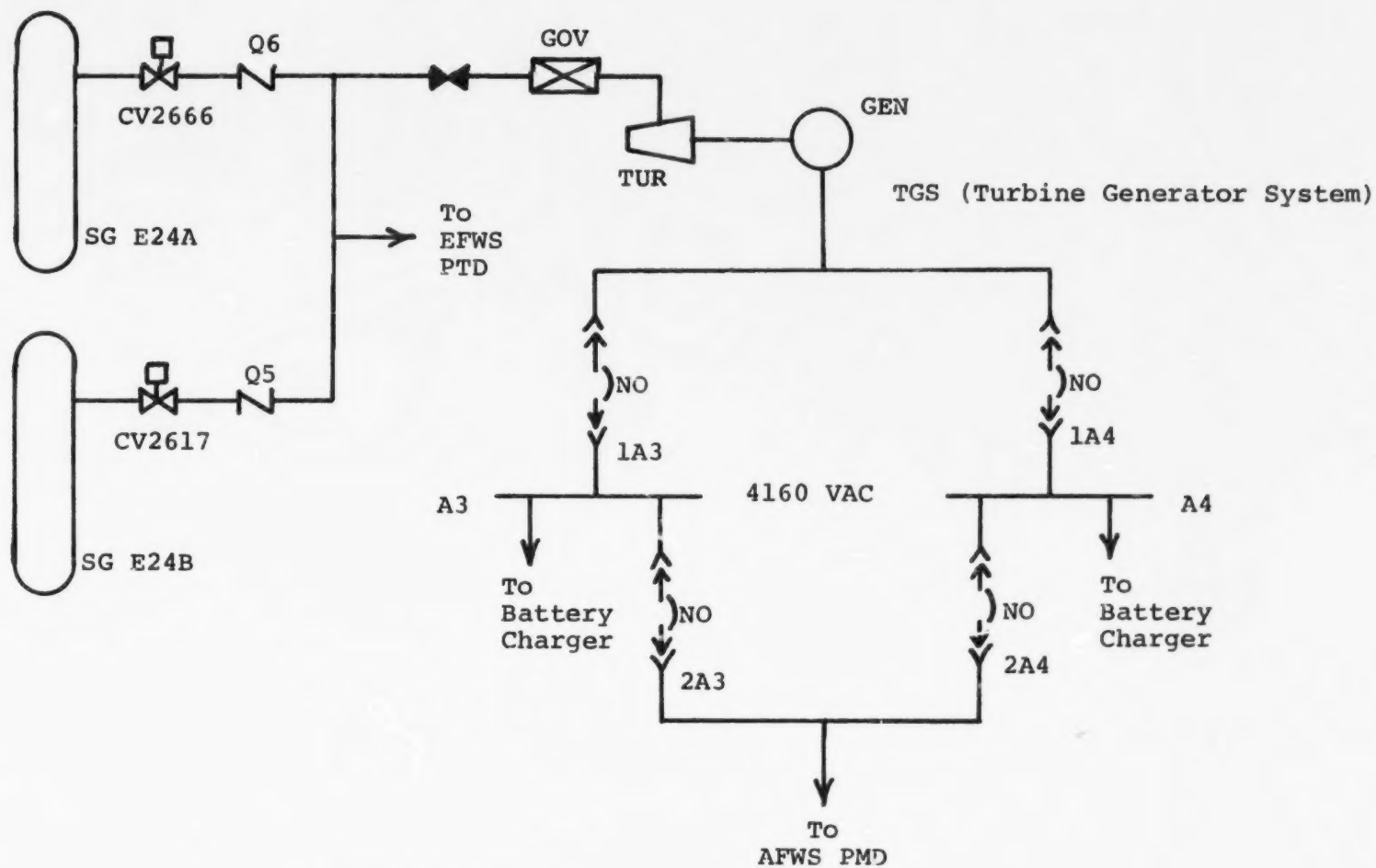


Figure 2.3 Modification for Vulnerability 5, ANO-1

INTERNAL VULNERABILITY 6: Failure of the Low Pressure Injection Pumps.

APPLICABLE CUT SETS:

LPIOP34A-PMD-LF*LPIOP34B-PMD-LF	2.42E-6
LPIOP34A-PMD-LF or LPIOP34B-PMD-LF*	<u>8.17E-6</u>
various other single failures	1.06E-5

Dominant accident sequences affected and estimated core melt probability contribution resulting from this vulnerability/cut sets

S ₃ MH ₁ H ₂ '	6.6E-6
T ₃ QH ₁ H ₂ '	3.30E-6
T ₂ MQH ₁ H ₂ '	4.66E-7
T ₃ MQH ₁ H ₂ '	<u>1.99E-7</u>
	1.06E-5

DESCRIPTION: Failure of both low pressure pumps or either pump and various other single failures leads to core melt.

MODIFICATION: Install third low pressure pump in parallel to the existing two pumps as shown in Figure 2.4a. This swing pump would eliminate UTM failure and significantly reduce the system failure by providing a backup pump for recovery of the system function given single or dual pump failures. Lube oil (Figure 2.4b) and room cooling (Figure 2.4c) are also required for this modification.

INTERNAL VULNERABILITY 7: Failure to manually initiate Feed and Bleed or high pressure recirculation.

APPLICABLE CUT SETS:

FNB-MANACT	1.98E-6
FNB-MANACT * various singles and doubles	2.82E-6
HPRS-MANACT * various singles	<u>3.01E-7</u>
	5.11E-6

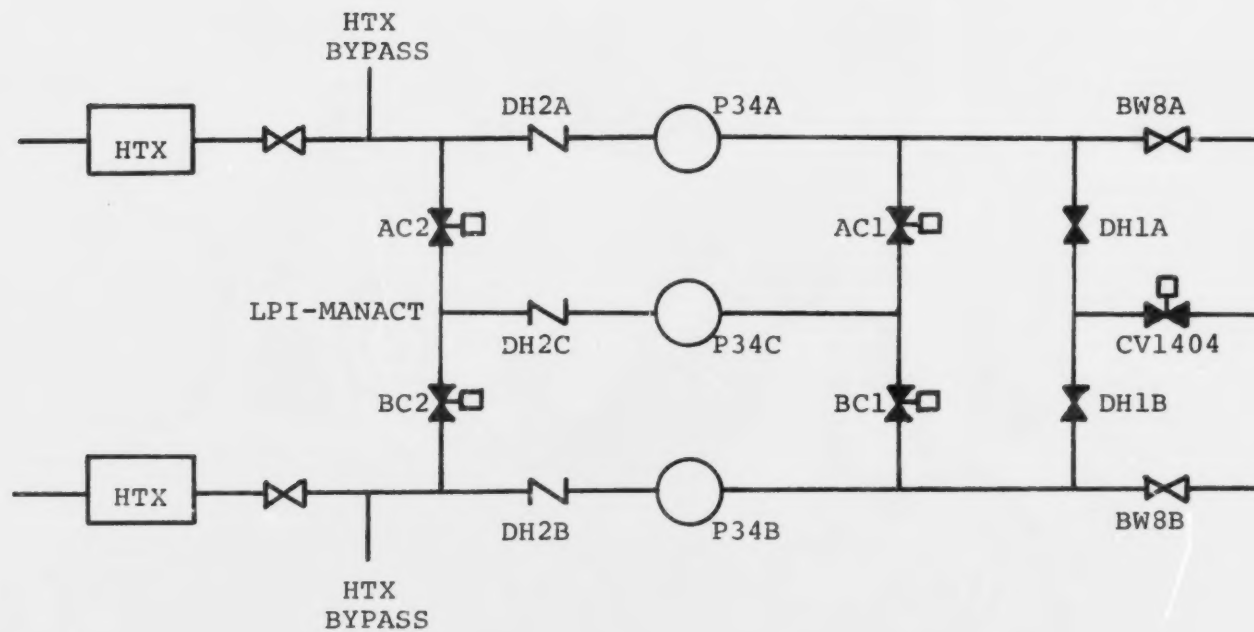


Figure 2.4a Modification for Vulnerability 6 - LPI Pumps, ANO-1

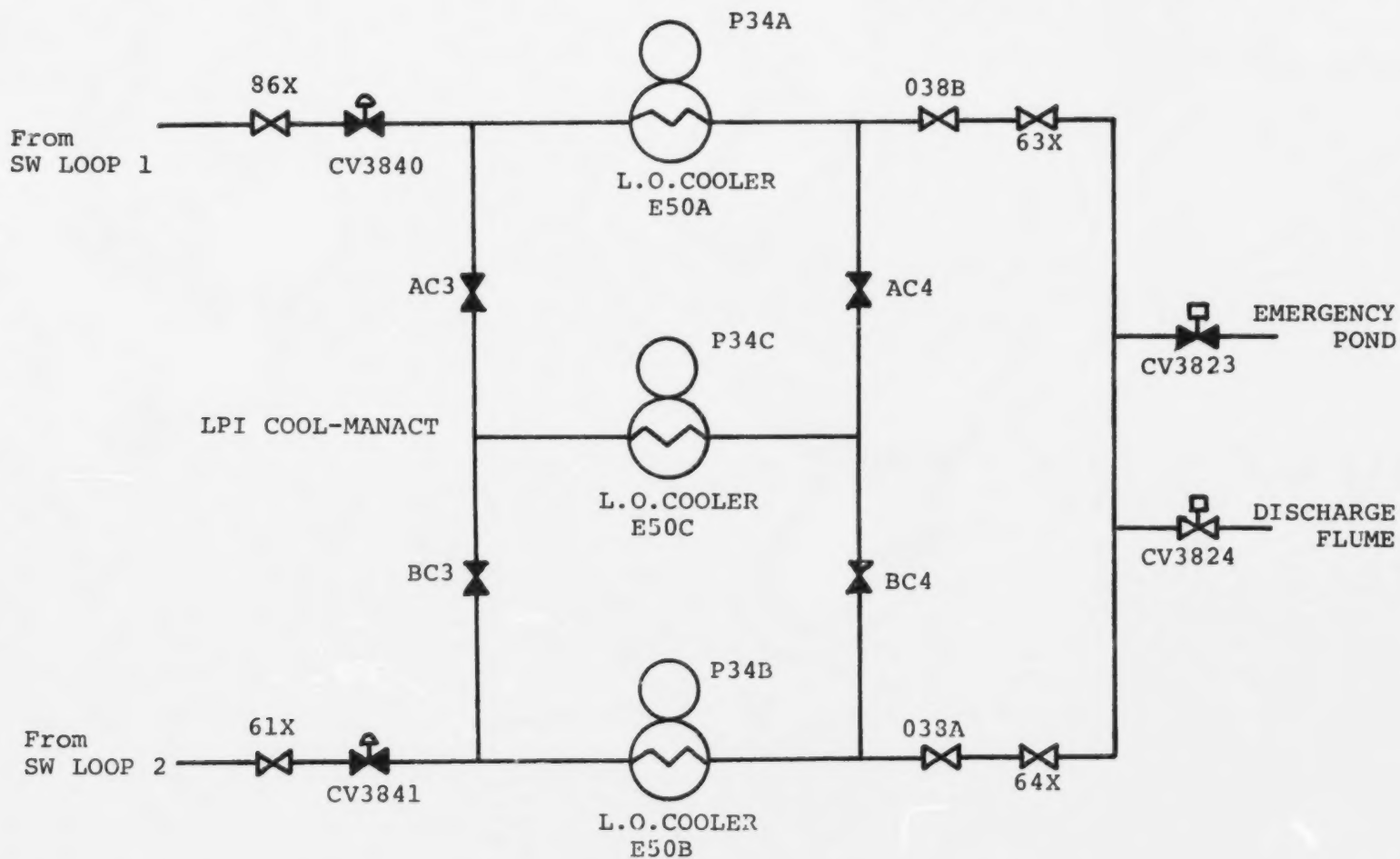


Figure 2.4b Modification for Vulnerability 6 - L. O. Cooling, ANO-1

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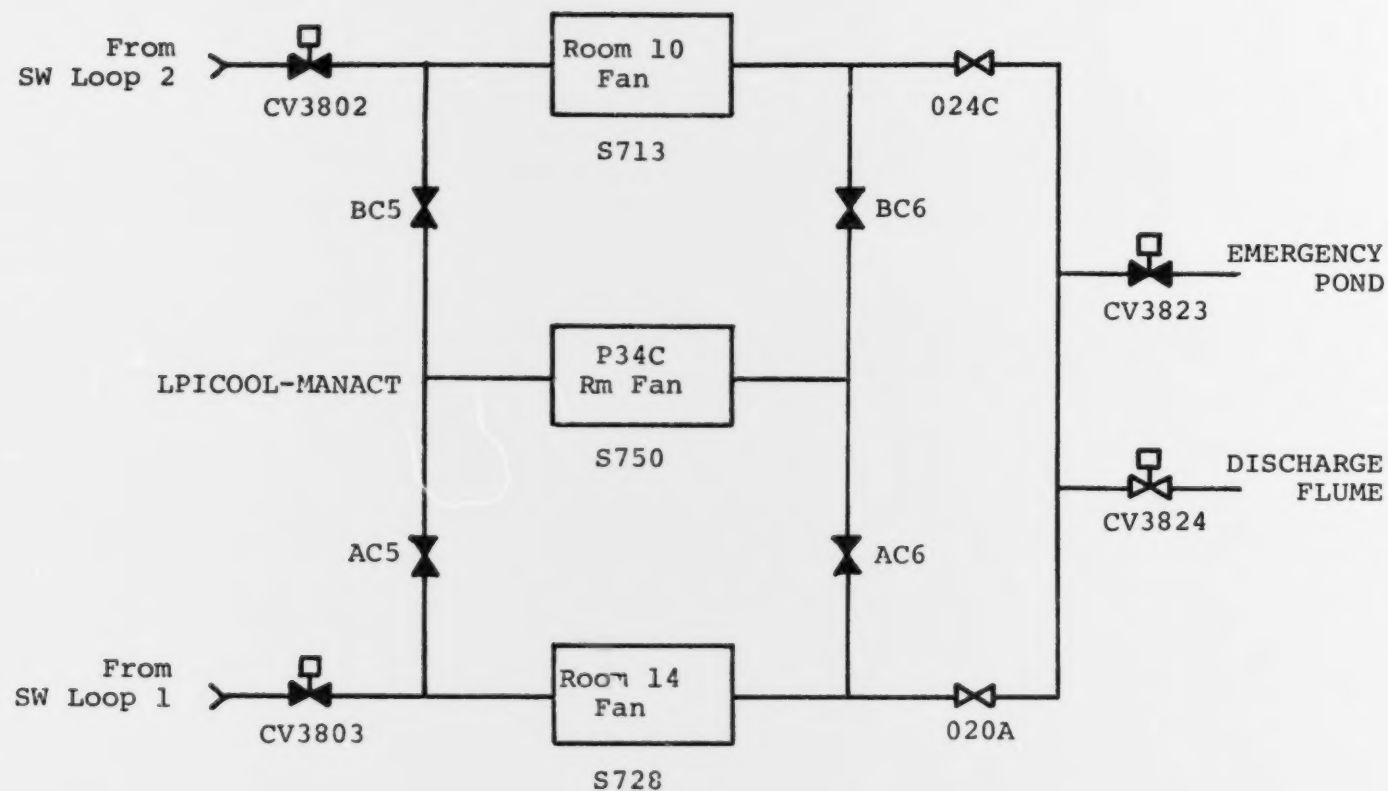


Figure 2.4c Modification for Vulnerability 6 - Room Cooling, ANO-1

Dominant accident sequences affected and estimated core melt probability contribution resulting from this vulnerability/cut sets

S ₃ MXE	1.98E-6
T ₂ MLE	1.35E-6
T ₃ MLE	6.26E-7
T ₄ MLE	5.68E-7
T ₅ MLE	2.79E-7
S ₃ MH ₁ H ₂ '	1.80E-7
T ₃ QH ₁ H ₂ '	<u>1.01E-7</u>
	5.11E-6

DESCRIPTION: In circumstances where feed and bleed is the backup mode of operation, failure of operator initiation leads to core melt. The operator is always required to transfer the high pressure system from injection to recirculation.

MODIFICATION: The operator error probability is 3E-3. Improved awareness and additional emphasis in operator training should reduce this vulnerability by a factor of 3. Thus there is no hardware modification proposed.

INTERNAL VULNERABILITY 8: Failure of the single manual discharge valve from the BWST or the BWST discharge valves leading to the safety system trains.

APPLICABLE CUT SETS:

LPIBW1X-XOC-LF	4.18E-6
LPI1407A-VCC-LF or LPI 1408B-VCC-LF*	
various singles	<u>1.74E-6</u>
	5.92E-6

Dominant accident sequences affected and estimated core melt probability contribution resulting from this vulnerability/cut sets

S ₃ MED ₂	2.47E-6
T ₃ QD ₁ D ₂	1.36E-6
S ₃ MH ₁ H ₂ '	1.11E-6
T ₃ QH ₁ H ₂ '	5.49E-7
T ₂ MQD ₁ D ₂	1.98E-7
S ₃ MXE	1.19E-7
Other	<u>1.11E-7</u>
	5.92E-6

DESCRIPTION: There is a single locked open manual valve on the discharge line from the BWST to the HP, LP, and Reactor Building spray systems. A local fault of this valve results in loss of suction to these safety systems. The LPIBWIX-XOC-UTM event was not modeled since in our judgment preventive maintenance would not be done on a single valve during operation. However, if maintenance were required for repairs due to a failure discovered during operations, the LF cut set probability could be as much as a factor of 2 greater to account for this.

After the single manual valve from the BWST, there are two valves connecting all the safety systems using this source. The failure of both these valves or either valve and various other failures can lead to core melt.

MODIFICATION: Install a parallel manual valve to improve the reliability of the BWST source to the safety injection systems. It is also proposed that a third discharge line be installed using air-operated valves maintained independently of the existing motor operated valves. These valves would receive the same signals as the corresponding trains but would provide an alternate path in the event of failures that cannot be manually corrected. Figure 2.5 addresses both valve problems described in this vulnerability.

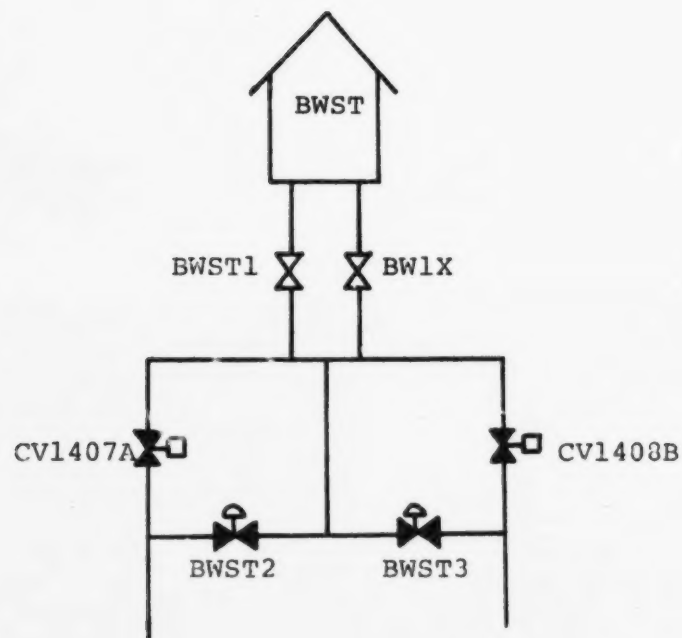


Figure 2.5 Modification for Vulnerability 8, ANO-1

2.4 Summary of Internal Analysis Modifications

There were eight potentially significant internal vulnerabilities identified with proposed modifications to six vulnerabilities.

Internal Modification 1 - Reconfigure the Auxiliary Feedwater motor driven pump to connected directly to the new CST and operate off Class 1E power.

Internal Modification 2 - No proposed modification

Internal Modification 3 - No proposed modification

Internal Modification 4 - Same as modification 1

Internal Modification 5 - Install a turbine driven generator to provide vital AC and DC, and instrumentation and control for the turbine driven EFW system pump.

Internal Modification 6 - Install third low pressure pump to the two existing pumps.

Internal Modification 7 - Improve operator awareness of the importance of certain manual operations and emphasize in training and procedures.

Internal Modification 8 - Add parallel discharge valves in the BWST output to the connecting safety systems.

3.0 SPECIAL EMERGENCY ANALYSES - METHODS AND RESULTS

Task Action Plan A-45 has been developed to assess the adequacy of decay heat removal in existing light water reactors in response to both "internal" and "external" challenges. While internal challenges are generally defined as those probabilistic random failures normally analyzed in a probabilistic risk assessment (PRA), few PRAs have considered the potential contribution to core melt probability due to the "external" events. Since many of these "external" challenges actually occur inside a plant (e.g., fire, turbine-generated missiles, pipe whip), we call them all special emergencies to avoid confusion.

The list of special emergencies that can be postulated to occur is quite extensive and a thorough analysis of each one would require resources beyond the scope of this program. However, reasonable rationale can be applied to exclude many special emergencies from consideration in this program. It should also be noted that combinations of special emergencies from different causes occurring simultaneously are not being analyzed here.

The PRA Procedures Guide (NUREG/CR-2300)⁷ lists four bases for excluding special emergencies from detailed analyses. These reasons are listed below:

1) The event is of equal or lesser damage potential than the events for which the plant has been designed. This requires an evaluation of plant design bases in order to estimate resistance to a particular external event. For example, it is established that safety-related structures designed for earthquake and tornado loadings can safely withstand a 1-psi peak positive incident overpressure from explosions (USNRC, 1978). Hence, if the PRA analyst demonstrates that the overpressure resulting from explosions at a source (e.g., railroad, highway, or industrial facility) cannot exceed 1 psi, these postulated explosions need not be considered. It is assumed that the conditional frequencies of failure of structures and components for overpressure of less than 1 psi are negligible given that the safety-related structures are designed for earthquake and tornado loadings. This screening criterion is not applicable to events like earthquakes, floods, and extreme winds since their hazard intensities could conceivably exceed the plant design bases.

2) The event has a significantly lower mean frequency of occurrence than other events with similar uncertainties and could not result in worse consequences than those events. For example, the PRA analyst may exclude an event whose mean frequency of occurrence is less than some small fraction of those for other events; the uncertainty in the frequency estimate for the excluded event is judged by the PRA analyst as not significantly influencing the total risk. Alternatively, the analyst may decide to compare event occurrence frequencies

at some high confidence level (e.g., 95 percent). After the total plant risk is estimated, the deleted external events may have to be reviewed to ascertain that a detailed assessment would not reveal them as significant contributors to the total plant risk.

3) The event cannot occur close enough to the plant to affect it. This is also a function of the magnitude of the event. Examples of such events are landslides, volcanic eruptions, earthquake-fault ruptures (seismic motion and its effects are treated under seismic events), and explosions.

4) The event is included in the definition of another event. For example, storm surges and seiches can be included in external flooding; the release of toxic gases from sources external to the plant is included in the effects of either pipeline accidents, industrial or military facility accidents, or transportation accidents.

For the purposes of TAP A-45, two additional arguments have been developed to exclude certain special emergencies from further analysis.

5) The event does not directly effect the decay heat removal systems and, therefore, the proposed protection does not involve a modification to the decay heat removal system.

6. The event is slow in developing and there is sufficient time to go to cold shutdown, to eliminate the source of the threat, or to prepare an acceptable alternative for the removal of decay heat.

An event deserving further discussion is aircraft impact. Due to the solid construction of plant structures, only a large aircraft would be capable of damaging decay heat removal systems. Aircraft impact is a low probability event. In addition, in the newer plants the separation (of redundant trains) used to satisfy fire, flood and sabotage concerns also serves to reduce the vulnerability to common mode failure due to an aircraft crash. In some older plants, redundant trains may not be as well separated, however, diverse systems (e.g. auxiliary feedwater and high pressure injection) usually are separated so that it is unlikely that a single event would disable all decay heat removal capability. Certainly, the addition of a dedicated decay heat removal system (in separate buildings) to reduce the core melt contribution from those special emergencies which are found to be important contributors to risk, would also reduce the aircraft vulnerability. The impact from aircraft crash debris is similar to tornado missiles, the resulting fire from the crash would be covered by the fire event, and any explosion would be similar to the explosion events. Therefore, aircraft accident analyses will not be pursued further.

Applying the foregoing rationale to the other postulated special emergencies gives the following results:

<u>Event</u>	<u>Applicable Rationale</u>	<u>Remarks</u>
Avalanche	3	Can be excluded for most sites in the United States
Biological Phenomena	6	Sufficient time to prevent damage
Coastal Erosion	4	Included in the effects of external flooding
Dam Failure	4	Included in the effects of external flooding
Drought	1,6	Excluded under the assumption that there are multiple sources of ultimate heat sink or that the ultimate heat sink is not affected by drought (e.g., cooling tower with adequately sized basin)
Explosions	1	Tornado missiles, tornadoes, and high winds govern
External Flooding	-	Requires detailed study
Extreme Winds and Tornadoes	-	Requires detailed study
Fire (internal)	-	Requires detailed study
Fog	1	Could, however, increase the frequency of man-made hazard involving surface vehicles or aircraft
Forest Fire	1	Fire cannot propagate to the site because the site is cleared
Frost/Freezing	1	Snow and ice govern
Hail	1	Other missiles govern
High Tide, High Lake Level or High River Stage	4,6	Included under external flooding

<u>Event</u>	<u>Applicable Rationale</u>	<u>Remarks</u>
High Summer Temperature	1	Ultimate heat sink is designed for at least 30 days of operation, taking into account evaporation, drift, seepage, and other water-loss mechanisms
Hurricane	-	Requires detailed study- partially included under external flooding: wind forces can be covered under extreme winds and tornadoes
Ice Cover	1,4	Ice blockage of river is partially included in flood
Internal Missiles	4	Included under turbine-generated missile
Industrial or Military Facility Accident	5	Included under explosions or toxic gas
Internal Flooding	-	Requires detailed study
Landslide	3	Can be excluded for most sites in the United States
Lightning		Requires study
Low Lake or River Water Level	1	Ultimate heat sink is designed for at least 30 days of operation, taking into account evaporation, drift, seepage, and other water-loss mechanisms
Low Winter Temperature	1,6	Thermal stresses and embrittlement are insignificant or covered by design codes and standards for plant design; generally, there is adequate warning of icing on the ultimate heat sink so that remedial action can be taken
Meteorite	2	All sites have approximately the same frequency of occurrence

<u>Event</u>	<u>Applicable Rationale</u>	<u>Remarks</u>
Pipeline Accident (gas, etc.)	4,5	Included under explosions and toxic gas
Pipe Whip	-	Requires detailed study
Intense Precipitation	4	Included under external and internal flooding
Release of Chemicals in Onsite Storage	4,5	Included under explosions and toxic gas
River Diversion	1,4,6	Considered in the evalua- tion of the ultimate heat sink; should diversion become a hazard, adequate storage is provided
Sabotage	-	Requires detailed study
Sandstorm	1,5	Included under tornadoes and winds
Seiche	4	Included under external flooding
Seismic Activity	-	Requires detailed study
Ship Collision	2	A loss of the crib house would not immediately threaten the core
Snow	1,4, 5,6	Snow melt causing river flooding is included under external flooding
Soil Shrink-Swell Consolidation	1	Site-suitability evaluation and site development for the plant are designed to preclude the effects of this hazard
Storm Surge	4	Included under external flooding
Transportation Accidents	4,5	Included under explosions and toxic gas
Tsunami	4	Included under external flooding and seismic events

<u>Event</u>	<u>Applicable Rationale</u>	<u>Remarks</u>
Toxic Gas	5	Does not threaten DHR systems
Turbine-Generated Missile	-	Requires detailed study
Volcanic Activity	3	Can be excluded for most sites in the United States
Waves	4	Included under external flooding

Therefore, this rather formidable list of special emergencies has been reduced on inspection to the following:

- External Flooding
- Extreme Winds and Tornadoes
- Fire (internal)
- Hurricane
- Internal Flooding
- Lightning
- Pipe Whip
- Sabotage
- Seismic Activity
- Turbine-Generated Missiles

The methods for analysis of pipe whip and turbine-generated missiles are beyond the resources of this program. These subjects are considered in the Safety Evaluation Reports (SERs) for individual plants and were considered again in the Standard Review Plan evaluations. Therefore, they will not be considered in the A-45 study. This leaves eight potential special emergencies for further consideration. Each of these special emergencies is addressed separately in this section. A brief explanation of the analysis methods, quantification of decay heat removal vulnerabilities to these events, and a description of potential modifications are given. The complete analysis details are provided in the referenced appendix.

It should be realized that the quantification of the special emergencies has large uncertainties in many cases. Since the state of the art is not very advanced in the analysis of many special emergencies, it was necessary to make assumptions in order to perform the calculations. These assumptions are documented in the following analysis summaries as well as in the appropriate appendices. Nonetheless, a residual risk may remain for such special emergencies as fire, flood, and earthquake.

3.1 Seismic Analysis

Seismic events have a wide range of factors which affect their threat to nuclear power plants. The location of the earthquake, the magnitude of the ground acceleration, the soil transfer of energy, the elevation of plant equipment, and the seismic equipment supports are only a few of the factors to be considered in an analysis of plant vulnerabilities. However, the fact that earthquakes in the United States have affected large portions of the country requires an analysis of the probability and uncertainties of a resulting core melt.

Successful completion of this program required a simplified seismic risk assessment methodology which could be applied to particular plants being studied in a far shorter time than previous PRA studies. This section describes the procedures developed for the simplified seismic analysis and presents the results of the risk quantification for the ANO-1 Nuclear Power Plant.

A more detailed description of the seismic analysis is presented in Appendix C of this report.

3.1.1 Analysis Methods

There are seven steps required for calculating the seismic risk at a nuclear power plant:

- 1) Determine the local earthquake hazard (Hazard Curve and Site Spectra).
- 2) Identify accident scenarios for the plant which lead to radioactive release (Initiating Events and Event Trees).
- 3) Determine failure modes for the plant safety and support systems (Fault Trees).
- 4) Determine fragilities (probabilistic failure criteria) for the important structures and components
- 5) Determine the responses (accelerations or forces) of all structures and components (for each earthquake level).
- 6) Compute the probability of core melt using the information from Steps 1 through 5.
- 7) Estimate uncertainty in the core melt probabilities.

Only the level of detail differentiates a simplified seismic analysis from a detailed seismic PRA. The seven steps of the TAP A-45 simplified seismic risk analysis procedure are described below.

Step 1 - Seismic Hazard Characterization -

- a) If a seismic hazard curve exists for the plant (e.g., from the Systematic Evaluation Program (SEP) program or from an existing PRA of a nearby plant) it is used.
- b) If no such curve exists, a hazard curve scaled to an exceedance probability of $2.5E-4/\text{yr}$ at the safe shutdown earthquake (SSE) will be used. The slope of the curve for higher peak ground acceleration (PGA) values is estimated from other hazard curves for the same broad seismological province.
- c) All plants west of the Rocky Mountains require site specific hazard curves due to the high levels of seismic activity, but most existing plants have hazard curves already available.

Step 2 - Initiating Events and Event Trees - The scope of TAP A-45 includes only the initiating events associated with small loss of coolant accident (SLOCA) and transient events. Two types of transients are being considered: those in which the power conversion system (PCS) is initially available (denoted Type EQ1 transients) and those in which the PCS is failed as a direct consequence of the initiating event (denoted Type EQ2 transients). The event trees derived for the internal event analyses for TAP A-45 are utilized.

The small LOCA initiating event frequencies are being computed based on the statistical distribution of small pipe failures computed as part of the NRC-sponsored Seismic Safety Margins Research Program (SSMRP) (Reference 2). The restriction to small LOCAs is not particularly limiting, because existing PRAs have shown that large and medium LOCAs are not major contributors to total risk of core damage.

The frequency of Type EQ2 transients is based on the probability of loss of offsite power (LOSP). This will always be the dominant cause of these transients (for the majority of plants for which LOSP causes loss of main feedwater). The Type EQ1 initiating event is computed from the condition that the sum of the initiating event probabilities considered must be unity. The hypothesis is that, given an earthquake of reasonable size (3.75 magnitude Richter was used in the SSMRP), at least one of the initiating events will occur.

Step 3 - Fault Trees - Simplified systems fault trees for a number of example plants have been developed as part of TAP A-45, (for random failures only). These fault trees will be used, although they require some modification to include basic events for seismic failure modes and re-solving the trees for pertinent cut sets to be included in the probabilistic calculations. Probabilistic culling is used in re-solving these trees to assure that important correlated failure modes are not lost.

Step 4 - Component and Structure Failure Descriptions -

Component seismic fragilities are obtained both from a generic fragility data base and from plant specific fragilities derived for components identified during a plant walkdown.

The generic data base of fragility functions for seismically-induced failures was originally developed as part of the SSMRP (Reference 3). Fragility functions for the generic categories were developed based on a combination of experimental data, design analysis reports, and an extensive expert opinion survey. The experimental data utilized in developing fragility curves were obtained from the results of the component manufacturer's qualification tests, independent testing lab failure data and data obtained from the extensive U.S. Corps of Engineers SAFEGUARD Sub-system Hardness Assurance Program. These data were then statistically combined with the expert opinion survey data to produce fragility curves for the generic component categories. Inasmuch as four years have passed since this data base was originally developed, the site-specific fragilities developed for some 20 PRAs performed in the interim were examined, and compared (Reference 4) against the original data base. This resulted in modifications of median fragility levels of 5 of the original 37 generic categories. For tipping and sliding of cabinets, and for dynamic analyses of tanks, graphical methods of analysis have been developed to greatly simplify the required fragility development.

Within the time scale prescribed, structural fragility development is not possible, and could not be included for TAP A-45. This is appropriate since it is the design requirements for decay heat removal systems which are being evaluated, and not the structural design. However, the approach being described here should prove widely useful in examining other generic and value/impact issues not directly concerned with adequacy of power plant structures.

Step 5 - Seismic Response of Structures and Components -

Building and component seismic responses are estimated from peak ground accelerations at several probability intervals on the hazard curve. Three basic aspects of seismic response--best estimates, variability, and correlation--are estimated. SSMRP Zion analysis results and simplified methods studies form the basis for assigning scaling, variability and correlation of responses.

In each case, SHAKE code (Reference 2) calculations are performed to assess the effect of the local soil column (if any) on the surface peak ground acceleration and soil-structure interactions. This permits an appropriate evaluation of the effects of non-homogeneous underlying soil conditions which can strongly affect the building responses.

Two situations are commonly encountered in developing

structural responses. First, for many early plants, only design floor spectra are available from the design reports. For this situation, the results of a detailed soil-structure interaction investigation (Reference 5) made as part of the SSMRP may be used to scale the computed design floor slab accelerations to best estimate values including the effects of the soil; and to estimate best estimate floor response spectra.

Second, one has the situation where fixed-base mass-spring or eigen-system descriptions are available for one or more buildings at the site. For this case, one can compute the floor slab accelerations using the CLASSI code (Reference 2). This code takes a fixed base eigen-system model of the structure and input-specified frequency dependent or independent soil impedances and computes the structural response (as well as variation in structural response if desired).

Variability in responses (floor and spectral accelerations) is assigned based on the SSMRP results. The recommended uncertainties (expressed as standard deviations of the logarithms of the responses, β) are shown below:

<u>Quantity</u>	<u>β random</u>
Peak Ground Acceleration	0.25
Floor Zero Period Acceleration	0.35
Floor Spectral Acceleration	0.45

Correlation between component failures is being included explicitly in TAP A-45. In computing the correlation between component failures (in order to quantify the cut-sets) it is necessary to consider correlations both in the responses and in the fragilities of each component. Inasmuch as there are no data as yet which show correlation between fragilities, the fragility correlations between like components are taken as zero and 1, and the possible effect of such correlation quantified in a sensitivity study. The correlation between responses is assigned according to a set of rules given in Reference 1. These are applied to both BWR and PWR plants.

Step 6 - Probabilistic Failure and Core Melt Calculations -

Given the input from the five steps above, the SEISIM (Reference 2) and SETS (Reference 6) codes are used to calculate the required output (probabilities of failure, core melt, etc.). Use of the SEISIM code for the final computation of accident sequences permits proper inclusion of seismically-induced correlation between component failures.

Step 7 - Estimate Uncertainties - Uncertainties in core damage estimates are being based entirely on upper and lower confidence limits for the hazard curve. Generic recommendations for such limits are given in Reference 1. This simplification is appropriate since previous seismic PRAs have

shown the hazard curve to contribute 70-90% of the total seismic risk uncertainty.

3.1.2 Seismic Vulnerability

The safe shutdown earthquake for ANO-1 has an acceleration of 0.20 g, however TAP A-45 is examining the vulnerability of decay heat removal systems to initiating events which are beyond the design basis. Therefore, probabilities of seismically induced core melt were calculated for earthquakes of four levels. For the ANO-1 site, the frequency of earthquakes of these magnitudes are estimated in Appendix C as follows:

<u>Earthquake Level</u>	<u>Frequency/yr</u>
.1 g	1.2E-3
.2 g	3.0E-4
.4 g	5.0E-5
.6 g	1.6E-5

An earthquake may initiate a core melt scenario by causing one of the following plant states: a small-small loss of coolant accident (S_3), a small loss of coolant accident (S_2), a transient in which the power conversion is initially available (T_3), and those transients in which the power conversion system has failed as a direct consequence of the initiating event (T_2). The frequency of Type T_2 transients is based on the probability of loss of offsite power since this will always be the dominant cause of these transients. While the probability of a small LOCA or a loss of offsite power increases as the earthquake level increases, the probability of a Type T_3 transient will decrease accordingly.

Since all of the plant systems feel the effects of an earthquake, the quantification of core melt scenarios involves calculating component fragilities for the various earthquake levels and using these values in the fault trees developed in the internal event analyses. Table 3.1 summarizes the probability of each of the seismically induced initial plant states for the varying earthquake levels. Included in this table are the resulting core melt probabilities for each earthquake level and initiating plant state. The total core melt probability is calculated to be approximately $7.3E-5/R$ yr.

The major contributions to core melt come in the .1 g to .4 g range and not for the larger earthquakes. The T_2 initiating event represents two thirds of the core melt probability.

3.1.3 Potential Modifications

The BWST is located in the yard (elevation 353 ft). It is predicted to have low seismic resistance, with the failure mode of shell buckling near the base and anchor bolt yielding. The proposed modification to this tank would be to add structural

Table 3.1 Initiating Plant State Probability and Core Melt Probability for Varying Earthquake Levels

Earth- quake Level	<u>S₃</u>		<u>S₂</u>	
	<u>Initiating State Probability</u>	<u>Core Melt Probability</u>	<u>Initiating State Probability</u>	<u>Core Melt Probability</u>
.1 g	9.0E-4	2.1E-6	1.0E-6	1.3E-7
.2 g	4.4E-3	5.3E-6	1.5E-3	2.2E-6
.4 g	8.0E-2	2.3E-6	3.8E-2	1.8E-6
.6 g	2.6E-1	6.9E-8	1.6E-1	4.9E-10

Earth- quake Level	<u>T₂</u>		<u>T₃</u>		<u>Total Core Melt Probability</u>
	<u>Initiating State Probability</u>	<u>Core Melt Probability</u>	<u>Initiating State Probability</u>	<u>Core Melt Probability</u>	
.1 g	4.8E-3	6.3E-6	9.9E-1	5.9E-6	1.4E-5
.2 g	2.6E-1	3.1E-5	7.4E-1	1.2E-6	4.0E-5
.4 g	8.8E-1	1.3E-5	1.0E-6	0.0	1.8E-5
.6 g	5.9E-1	3.0E-7	1.0E-6	1.4E-7	5.2E-7
					7.3E-5

stiffening so as to increase the median ground acceleration level at failure to approximately 4 times the SSE which would significantly reduce its contribution to the probability of core damage.

The second proposed modification would be to anchor the 4160 VAC and the 480 VAC switchgear to the floor, and also, cabinet to cabinet.

Table 3.2 presents the recalculated core melt probabilities for the varying earthquake levels and initiating plant states if all of these modifications were implemented. The total core melt probability with these modifications incorporated would be approximately $8.8\text{E}-6/\text{R yr}$.

3.1.4 Summary and Conclusions

Two cases of the seismic risk at ANO-1 have been analyzed, the base case which includes a number of potentially vulnerable site-specific items, and the case with all site-specific vulnerable components upgraded. In both cases credit was taken for feed and bleed cooling and secondary blowdown.

The base case has a $7.30\text{E}-5$ per year probability of core damage, and an estimated exposure of 4.90 man-rem/year. The dominant base case sequence is a transient with LOSP (~ 70%) or with PCS lost (~ 10%) involving failure of the EFWS and HPIS. The dominant contributors to this sequence are failures of different combinations of two electrical busses and either the CST or EFW motor-operated valve 2802. The 4160 VAC and 480 VAC busses were identified as potential site-specific vulnerabilities at ANO-1. The remaining percentage (20%) of core melt frequency is made up of various small-small LOCA and small LOCA initiated sequences. A small-small LOCA initiator with early failures of HPIS and LPIS is a dominant (10%) contributor to core melt within this group. Dominant cut sets in this sequence are combinations of two electrical bus failures.

In the second case analyzed, the BWST, 4160 VAC and 480 VAC busses are assumed to be upgraded so that their seismic vulnerabilities are no longer site specific. In this case, the frequency of core damage drops to $8.83\text{E}-6$ per year, and the exposure drops to 0.578 man-rem/year, nearly an 88% decrease in risk. The two most dominant sequences are again $T_2(2)$ at 44% and $S_3(3)$ at 13%. Four other sequences [$S_3(3)$, $S_2(3)$, $S_3(3)$, and $S_2(4)$] contribute between 10% to 7% each. Blackout type cut sets dominate the transient and $S_3(7)$ sequences; while blackout cut sets, or battery failures paired with either a SWS valve or ventilation system fault dominate the $S_3(3)$, $S_2(3)$, $S_3(4)$, and $S_2(4)$ sequences.

In the base case, the dominant failure mode was different combinations of electrical bus failures with or without another event to fail the EFWS. With modifications these types of cut

Table 3.2 Initiating Plant State Probability and Core Melt Probability With Modifications Implemented

Earth- quake Level	S_3		S_2	
	Initiating State	Core Melt	Initiating State	Core Melt
	<u>Probability</u>	<u>Probability</u>	<u>Probability</u>	<u>Probability</u>
.1 g	9.0E-4	1.2E-7	1.0E-6	9.9E-9
.2 g	4.4E-3	9.3E-7	1.5E-3	6.0E-7
.4 g	8.0E-2	2.1E-6	3.8E-2	1.1E-6
.6 g	2.6E-1	3.5E-9	1.6E-1	3.3E-10

Earth- quake Level	T_2		T_3		Total Core Melt <u>Probability</u>
	Initiating State	Core Melt	Initiating State	Core Melt	
	<u>Probability</u>	<u>Probability</u>	<u>Probability</u>	<u>Probability</u>	
.1 g	4.8E-3	1.0E-7	9.9E-1	1.1E-7	3.5E-7
.2 g	2.6E-1	1.2E-6	7.4E-1	8.7E-9	2.7E-6
.4 g	8.8E-1	2.6E-6	1.0E-6	0.0	5.8E-6
.6 g	5.9E-1	4.3E-9	1.0E-6	2.0E-9	<u>1.0E-8</u>
					8.8E-6

sets no longer dominated. The modified case was dominated by common mode failures of either pumps or valves, and different combinations of support system failures such as batteries, ventilation, and service water.

3.2 Fire Analysis

Based on plant operating experience over the last 20 years, it has been observed that nuclear power plants will probably have three to four significant fires over their operating lifetimes (i.e., on the order of one fire every ten years for a frequency of .1 fire/R yr). Previous probabilistic risk assessments have shown that fires are a significant contributor to the overall core melt probability, contributing from 7% to 50% of the total core melt probability.

Large strides have been made in the fire protection for nuclear power plants in recent years. In particular, 10 CFR 50, Appendix R and Branch Technical Position 9.5-1 have led to improved fire protection measures at nuclear power plants. However, even with these improvements, there remains a residual risk from fires which has not yet been quantified with the exception of those plants that have performed fire-related PRAs. The purpose of this section is to quantify the risks due to fires at the ANO-1 Nuclear Power Plant and to evaluate the value of potential modifications.

A more detailed description of the fire analysis is provided in Appendix D of this report.

3.2.1 Analysis Methods

This analysis will evaluate the residual risk from fires that could affect decay heat removal systems. In doing so, existing procedures used in published fire PRAs will be employed, but simplifying assumptions will be made to allow the analysis to proceed without requiring the level of detail required for these other fire PRAs.

Most of the simplifying assumptions are based on insights gained from the previous PRAs or from the historical fire data base. These assumptions are as follows:

a) Only fires in a single area (room) will be considered since only single areas have been seen to be significant contributors to core melt probability. Because no room to room issues are being considered, barriers between adjacent areas are assumed to remain intact during a fire.

b) Fire frequency and suppression reliability will be based on available industry data. The fire frequencies used will be the mean value of generic data collected from fire PRAs. Suppression system data is based on American Nuclear Insurers data and on the HTGR fire methodology PRA.

c) Two transient combustible exposure fires are assumed to bound all transient and electrically-initiated fires. A trash can size (30 gallons) container of refuse with an energy output of approximately 500KW for 15 minutes is one exposure fire used in the computer calculations. The other combustible is a 10 gallon spill of acetone spread over a square area with sides of 2 meters. This fire results in an intensity of 4.67MW for approximately three minutes. Electrically-initiated fires are not being explicitly evaluated, but it is assumed that the exposure fires postulated for this analysis bound any transient or electrically-initiated fire that might occur at a plant. The worst case geometries are used in assessing fire growth and spread and the fire code COMPBRN is used to determine if the cables or components of interest would be damaged by the fire.

d) The ANO-1 transient event trees will be used to determine the systems components, from a success-oriented outlook, that need to be considered. Transient event trees are used in this analysis since it is assumed that a fire occurs independently of any initiating event and the operators will always insure that at least a transient will occur once they know that there is a fire (i.e. scram the reactor).

e) Components affected by a fire are electrical/ active components (e.g., cables, motors, switchgear, buses). Only temperature will be used to evaluate a components vulnerability to a fire. Passive components, such as pipes and manual valves will be assumed to be unaffected by a fire.

f) Random failures of those systems not directly affected by the fire at a particular location will be considered if applicable.

g) Offsite power is assumed to be available during a fire unless the fire is capable of causing a loss-of-offsite power.

h) Fire growth and spread and suppression of fires are treated as independent of each other.

i) No explicit core melt timing is evaluated.

j) Where exact locations of cables cannot be accurately determined, assumptions as to their locations are made.

k) Credit is given in the calculations for use of radiant energy shields, conduit, solid bottom cable trays, and completely enclosed trays.

The basic approach used in determining the risk from a fire uses the following four basic steps:

a) Initial screening for potentially important fire areas.

- b) Plant visit for verification.
- c) Quantification of risk.
- d) Proposed fixes to reduce the fire risk in affected areas.

The initial screening determines which areas of the plant contain enough front line or support systems in their boundaries such that fire damage could invalidate all of the success branches on the transient event trees. These transient event trees identify the success criteria of the frontline systems or combinations of systems that prevent a core melt scenario. The internal systems analysis is also used as a starting point for the fire analysis to identify the success criteria for such support systems as cooling systems and the AC/DC power requirements. Plant schematics and piping diagrams supplement this information where appropriate. Available documentation then allows screening of fire areas to determine the small set of areas where a single fire could lead to a core melt.

The plant is then visited to see first hand what the physical arrangements are in each of the vital areas. The locations of major fuel sources and some estimate of the coordinates of components and cables are obtained.

In the quantification step, the deterministic fire code COMPBRN is used to ascertain if an exposure fire could damage the components/cables of interest in the area. An event tree is utilized to probabilistically estimate the risk of core melt from a fire in those areas not eliminated by the code calculations. The basic elements of the event tree are as follows:

- a) Fire occurrence frequency for the area of interest.
- b) Suppression system probability (both for automatic and manual suppression).
- c) Operator error and random failure of unaffected systems from a remote shutdown location.

Based on the quantification steps, engineering modifications are proposed for those areas which have a fire-induced core melt probability contribution of greater than $1.0E-6$. For any proposed modification, a new core melt probability is calculated by modifying the event tree to reflect the fix. This new core melt probability indicates the "value" of a modification in terms of core melt reduction.

3.2.2 Fire Vulnerability

Using the transient event trees, the following systems and success criteria were identified as crucial to preventing core damage at ANO-1:

- a) Power Conversion System: 1 of 2 trains and 1 of 3 condensate boost pumps
- b) Emergency Feedwater System: 1 of 1 motor driven pumps or 1 of 1 turbine driven pump
- c) High Pressure Injection System: 1 of 3 trains
- d) Auxiliary Feedwater System: 1 of 1 pumps and 1 of 3 condensate boost pumps

The support systems required for these frontline systems are:

- a) Service Water: 1 of 3 SW pumps
- b) AC Power: Normal offsite power or 1 of 2 diesel generators
- c) DC Power: 1 of 3 battery chargers or 1 of 2 station batteries.

The areas identified in the utilities Appendix R submitted and the plant visit are those chosen for analysis and listed below:

1. Auxiliary Building, Elevation 317'
2. Portions of the Auxiliary and Turbine Buildings
3. Auxiliary Building, Elevation 335'
4. "Green" Diesel Generator Room
5. South Switchgear Room
6. South Battery Room
7. Control Room (CR) and Cable Spreading Room (CSR)
8. "Red" Diesel Generator Room
9. North Switchgear Room and North Electrical Penetration Room
10. Containment
11. Intake Structure
12. Yard Area Manholes 1MH04 and 1MH06
13. Yard Area Manholes 1MH09 and 1MH10

The only area remaining after the initial screening is Area G containing the CR (Zone 129-F) and the CSR (Zone 97-R).

Zone 129-F - Control Room

The control room (CR) contains circuitry for all safe shutdown systems. An alternate shutdown is provided independent of the control room and an automatic Halon system is provided to protect the false floor and ceiling areas. The emergency system panels are located close together, but are separated by a fire barrier between panels C-16 and C-18. Further, the feedwater and condensate system panels are located separate from the emergency systems. Consistent with the original assumptions, this control room configuration was not analyzed further.

Zone 97-R - Cable Spreading Room

The cable spreading room (CSR) contains circuitry for all safe shutdown systems. An automatic water suppression system and a remote shutdown are provided for this area. The cables in this room are located in cable trays throughout the room at various elevations from the floor to the ceiling. Some cable trays also cross one another at various elevations. Because the exact locations of redundant cabling in the CSR are not known, assumptions concerning the location of certain cables are necessary. The location of equipment in the control room gives an indication of where the cables are located in the CSR. Based on the control room layout, a horizontal distance of 2.13 meters (7 feet) was estimated between the emergency systems cables and the PCS cables in the CSR. Consequently, these estimated distances were used in COMPBRN calculations. The COMPBRN calculations indicate that for the acetone fire, the redundant cables would ignite approximately two minutes into the fire.

The generic probability of a CSR fire is $7.0\text{E}-4/\text{R yr}$ (median) or $6.7\text{E}-03/\text{R yr}$ (mean). Using the mean value for fires in the CSR yields a room probability for fire of $6.7\text{E}-03/\text{R yr}$.

A water suppression system is installed in the CSR; the generic unreliability of this system is 0.04/demand. In the event of automatic suppression failure, manual suppression may still succeed. The generic probability of nonsuppression in five minutes (the approximate time to damage of redundant trains) is 0.4/demand.

The probability of failure to correctly align and start the primary remote shutdown system (EPW) is taken as 0.1/demand. This value is based on Seabrook data for the conditional frequency that the operators will not do the right thing at auxiliary shutdown panels. Once the system is running, the probability of failure is taken as 0.01/demand. In cases where the operator has failed to start the primary remote shutdown system or the primary remote shutdown system has failed, a backup shutdown method is available using the HPIS system in a

"feed-and-bleed" mode of operation. The probability of successfully aligning and starting the secondary shutdown method, given failure of the primary shutdown system, is taken to be 0.5. Once the backup system is started, the probability of random failures in the system is taken as 0.01. The resulting core melt probability from fires in the CSR is $5.8E-06/R$ yr.

3.2.3 Potential Modifications

The proposed modification is to install redundant detection, piping, and valving for the current water suppression system to improve its reliability. The redundant equipment would require independent power and would require valving and piping which would minimize the potential for common mode failures resulting from improperly positioning any single valve (especially manual valves). The probability of failure of the modified automatic suppression system is taken as the product of the independent probabilities plus the probability of a common mode failure. The common mode failure probability is estimated as one order of magnitude less than the independent system reliability, consistent with a probabilistic risk assessment practice when specific data is not available. Thus, the probability of failure of automatic suppression is $(0.04 \times 0.04) + 0.004 = 0.0056$. Using this value results in a core melt probability of $8.1E-07/R$ yr.

Details of this fire analysis are given in Appendix D along with the analysis event tree methodology.

3.3 Internal Flood Analysis

Operating experience at nuclear power plants has shown that the potential for damage to safety-related equipment as a result of internal flooding is of concern. Examples of experienced internal flooding initiators include operators overfilling water tanks, in-leakage of ground water, hose and weld ruptures, pump seal leaks, improper maintenance procedure, and various other circumstances. Several of these events resulted in damage to high pressure coolant injection systems, reactor building spray pumps, service water pumps, emergency feedwater pumps, diesel generator control cabinets and other redundant and diverse safety-related components. Although to date, none of these floodings have led to a serious accident, there is still a risk that needs to be evaluated to determine what the chances are that an internal flood, despite a conscious effort to design the plant to withstand this threat, could lead to a core melt scenario. The TAP A-45 program specifically addresses these concerns including the vulnerability of the decay heat removal systems to equipment damage due to spray from ruptured water piping and the sections that follow will represent the methodology and assumptions employed to carry out these analyses. The ANO-1 vulnerabilities to this special emergency will then be quantified and modifications evaluated. More details regarding this analysis are provided in Appendix E.

3.3.1 Analysis Methods

The methods used to analyze internal flooding in TAP A-45 follow the existing procedures employed in published PRAs. However, since a full-scope PRA is beyond the resources of this program, a number of simplifying assumptions have been made. Most of these assumptions are based on insights derived from previous PRAs or by consideration of the physical constraints of a flood in a building. The major assumptions are as follows:

a) Internal flood frequencies, component damage thresholds, and physical room considerations will be based on previous industry wide data bases. The flood sizes have been grouped according to the flow rate from a flood source and estimates have been made for the room drainage and equipment volume based on the Seabrook PRA assumptions.

b) The ANO-1 transient event trees are used to determine the systems/components, from a success oriented outlook, that need to be considered. Transient event trees are used in this analysis since it is assumed that an internal flood occurs independently of any initiating event and the operators will always insure that at least a transient will occur once they know that there is a flood (i.e. scram the reactor). LOCA event trees will not be examined since pipe ruptures have already been considered in these trees.

c) Front line mitigation systems (such as high and low pressure injection) were not considered as flood sources. This simplifying assumption is based on the fact that these front line systems, 1) have been designed to higher standards, 2) have controlled chemistry, 3) in general do not see large temperature gradients, and 4) have been involved in few catastrophic failures.

d) Flood barriers maintain their structural integrity.

e) Flooding via drains backing up into other areas was not considered.

f) Only single flood sources were analyzed.

g) Components will fail from submergence in water to a critical point or from direct spray from the flood source. It is assumed that switchgear and buses will fail if they are standing in six inches of water or more. An electric motor will be assumed to fail when the water level reaches the bottom of the motor casing.

h) Random failures of those systems not directly affected by the internal flood at a particular location will be considered if applicable.

i) Offsite power is assumed to be available during an internal flood unless the flood is capable of causing a loss of offsite power.

j) No explicit core melt timing is evaluated.

The approach used in determining the risk due to internal flooding at the ANO-1 Nuclear Power Plant uses the following four basic steps:

a) Initial screening for potentially important internal flood levels.

b) Plant visit for verification.

c) Quantification.

d) Proposed fixes to reduce the risk due to internal flood in affected areas.

The initial screening involves using the transient event trees to identify the front line and support systems needed to prevent core melt and their success criteria. Once the systems of interest are identified, all available documentation is analyzed to determine those areas where enough components could be disabled by an internal flood so as to cause core damage.

A plant visit is then conducted to see the physical arrangement of equipment, verify the accuracy of documentation, and obtain the plant location coordinates of vital components.

The step of quantifying the contribution to core melt begins with examining whether a flood source could damage the components of interest and the time available for operator mitigation of the event. With the flowrate, room volume, and capacity for water removal known, the time to damage of components may be calculated based on the water depth in the room and the assumptions governing component damage thresholds. Spray is considered when a flood initiates in a room and the critical components are in a direct line of sight with the flood source. An event tree is then constructed to quantify the area flood damage scenario following the following steps:

a) Internal flood occurrence frequency for the area.

b) Probability of detection of the flood.

c) Isolation potential of the flood source.

d) Operator error and random failures of unaffected systems.

Based on the quantification steps, engineering fixes are proposed for these areas which have an internal-flood-induced core melt probability contribution of greater than $1.0E-6$. For any proposed fix, a modified core melt probability is then calculated (by modifying the event trees to reflect the fix) to indicate how much the core melt probability could be reduced.

3.3.2 Internal Flood Vulnerability

The ANO-1 transient event trees were used to identify the decay heat removal system combinations which are needed to prevent core damage. These systems and their success criteria are summarized below:

- a) Power Conversion System (PCS): 1 of 2 trains and 1 of 3 condensate boost pumps
- b) Emergency Feedwater System: 1 of 1 motor driven pumps or 1 of 1 turbine driven pumps
- c) High Pressure Injection System: 1 of 3 trains
- d) Auxiliary Feedwater System: 1 of 1 pumps and 1 of 3 condensate boost pumps

Because the success criteria only requires 1 train or pump from each system to remove decay heat, then in order to invalidate the success criteria locations must be found in the plant where all of the redundant systems can be damaged by an internal flood. In addition, the support systems needed for successful operation of the front line systems are as follows:

Service water: 1 of 3 SW pumps

AC power: Normal offsite power or 1 of 2 diesel generators

DC power: 1 of 3 battery chargers or 1 of 2 station batteries

A review of these critical systems, in conjunction with Appendix B submittals, produced the following list of potential component locations:

1. Area G - Control Room and Cable Spreading Room
2. Area H - "Red" Diesel Generator Room
3. Area I - North Switchgear Room and North Electrical Penetration Room
4. Area J - Containment
5. Intake Structure
6. Yard Area Manholes 1MH04 and 1MH06
7. Yard Area Manholes 1MH09 and 1MH10

8. Elevation 317'0"
9. Elevation 335'0"
10. Elevation 354'0"
11. Elevation 368'0" - 374'6"
12. Elevation 386'0"

Based on the initial screening, no individual areas were identified that might be vulnerable to internal flooding.

Only one level, elevation 335'0", was identified as having a sufficient amount of equipment that safe shutdown could be prevented. All other levels either do not have a sufficient amount of equipment or drainage to lower levels through hatches or stairwells would prevent the redundant equipment from being damaged. Elevation 335'0" is located below grade and is the elevation where water would accumulate if internal flooding were to occur at higher elevations.

Internal flooding of all of this level has the possibility of damaging the PCS, HPIS, AFW, and EFW. The PCS and AFW could be disabled because all the condensate booster pumps are located at this elevation in the turbine building. The EFW and HPIS could be disabled because the respective pumps are all located at this elevation in the auxiliary building. An internal flood would thus have to affect both the auxiliary building and the turbine building. These two buildings are separated by fire walls at this level, but sources from above this level could possibly flood both areas simultaneously. The level that would have to be flooded is approximately 4200 m² (45000 ft²). An estimated water level of 0.46 meters (1.5 feet) would be required to damage all the pumps.

Even a complete emptying of the borated water storage tank (approximately 1,440,000 liters) onto this elevation would not provide the required water volume. Also, pipe breaks of the sizes postulated for this analysis would require hours of unmitigated flooding before they could even approach the required volume of water. Because of these considerations, flooding of both buildings of this elevation to a sufficient water depth is not deemed feasible. A flood in the turbine building would still leave several automatic safety systems available and a flood in the auxiliary building would leave the PCS available.

3.3.3 Potential Modifications

No locations were identified at Arkansas Nuclear One which are vulnerable to internal-flood-induced core damage accidents with a probability of greater than 1.0E-06/R yr. Much of the active safe shutdown equipment is located on the upper levels which are very large. Electrical distribution equipment is located in individual areas which have significant drainage paths available in the immediate vicinity of the areas. Finally, the control

room is protected by its location, its continuous manning, and drain paths immediately outside the area. Thus no modifications are proposed for the ANO-1 plant related to internal flooding.

3.4 External Flood Analysis

Sources of potential flooding external to a plant are numerous for most plant sites because nearly all plants are located near large bodies of water. River bank overflow, snow melting, heavy precipitation, storm seiches, upstream dam failure, and wave runup are examples of potentially large flood water sources. Flood waters can cause damage or failure to components located in the plant yard as well as damage to equipment inside buildings which may be penetrated by the water. In addition, the buildings themselves may not be designed to withstand the hydrostatic forces of flood waters, resulting in catastrophic structural failures.

However, unlike other natural hazards such as earthquakes and extreme winds, external flooding may not represent a generic hazard to all plant sites since there may be topographic or other physical conditions which preclude site flooding, or reduce the frequency of occurrence. This is due in part to the fact that the design basis floods for newer nuclear plants are generally conservative. For older plants such as ANO-1, the uncertainty in design floods was not considered at the time of construction and, therefore, the likelihood of extreme floods and their contribution to plant risk has not previously been calculated.

This section will analyze the potential sources of external floods for the ANO-1 Nuclear Power Plant and then quantify the contribution to core melt probability.

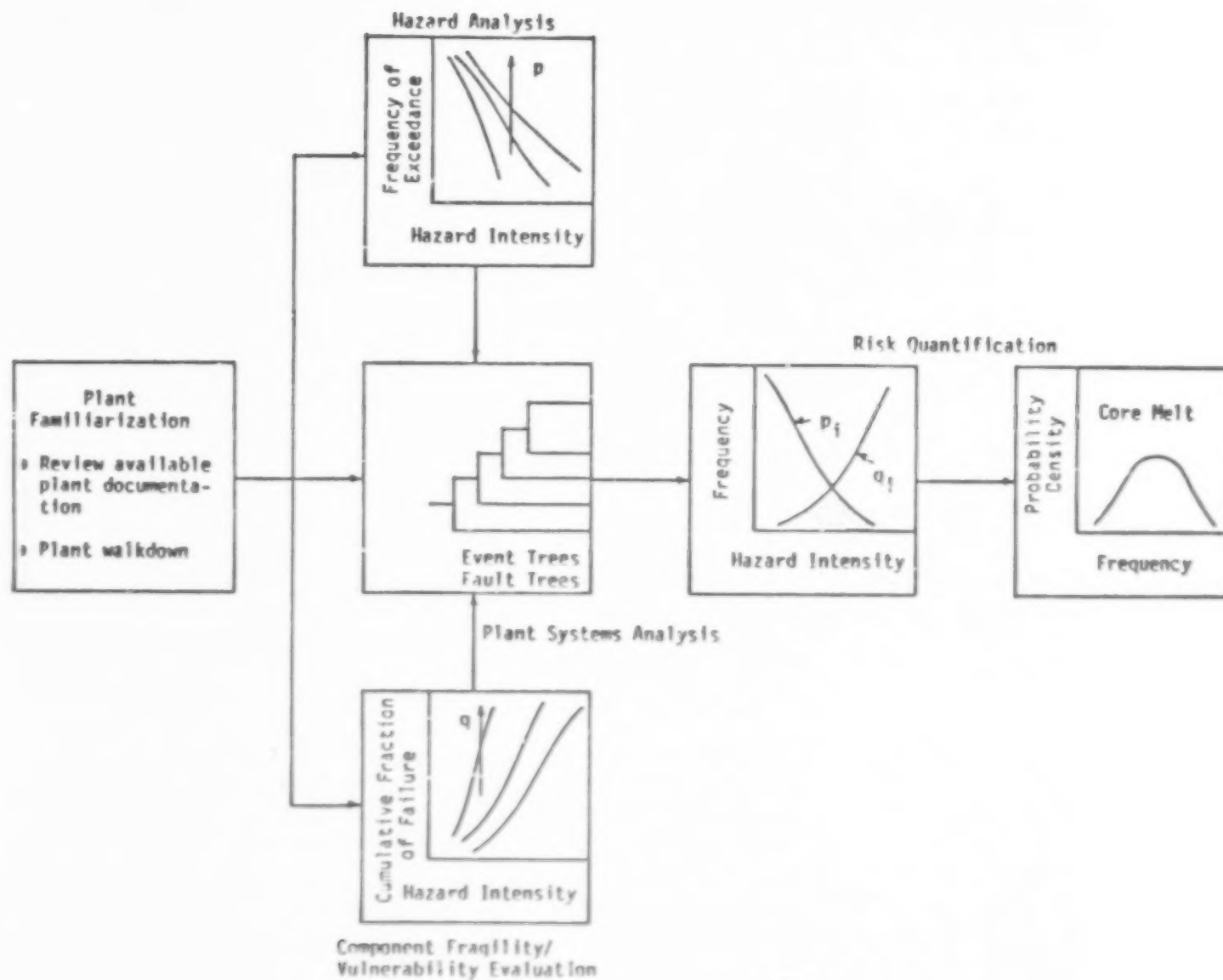
A more detailed description of this analysis is provided in Appendix F of this report.

3.4.1 Analysis Methods

It is beyond the scope of the TAP A-45 program to perform an in-depth probabilistic risk assessment. Instead, the objective is to conduct an analysis that quantifies the significant threats to a plant. However, the procedure used to perform the flood analysis embodies the basic philosophy of a full scope probabilistic risk assessment.

To evaluate the probability of plant damage due to external flood events, the following tasks are performed: 1) plant familiarization, 2) hazard analysis, 3) fragility/vulnerability evaluation, 4) systems analysis, and 5) risk quantification. Figure 3.1 shows the relationship of these elements.

The steps in the analysis are summarized below:

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Figure 3.1 Risk Analysis Flow Chart

Plant Familiarization - The purpose of this task is to gather information on the occurrence of external floods hazards at a plant and on the vulnerability of the plant structures and equipment to flooding. Regional studies performed by the Army Corps of Engineers are combined with plant site studies to evaluate the flood frequency. In assessing the reliability of shutdown decay heat removal systems, a review of available documentation (i.e., FSAR, plant drawings and topical reports) and a walkdown of the plant are performed. Safety related components are visually inspected to verify locations and to analyze for possible flooding via all possible pathways.

Hazard Analysis - In the case of external flooding, the hazard analysis is conducted in two steps. First, a screening of the potential sources of flooding (i.e., upstream dams, local precipitation, storm surge, etc.) in the vicinity of the plant is conducted. For sources of flooding that can impact the plant site, an analysis is conducted to evaluate the frequency of occurrence of increasing levels of flood intensity (i.e., depth of inundation, hydrodynamic forces, etc.). An uncertainty analysis is also performed to assess the modeling or engineering uncertainty in estimating the frequency of occurrence.

Component Fragility Evaluation - For structures and equipment items vulnerable to flood hazards the fraction of failure defined as a function of a flood hazard parameter (i.e., depth of inundation) or fragility curve, is assessed. This is done by 1) identifying the flood protection devices, 2) estimating the depth of submergence necessary to fail a component, 3) determining modes of flooding a room, and 4) transforming local fragility values or critical depth estimates to a site flood hazard characterization. The uncertainties in the modeling used for this evaluation are also considered.

Systems Analysis - The decay heat removal systems analysis is performed by developing event trees and fault trees with an external flooding event as the initiator. The system success criteria are examined for common mode failure due to flooding which could lead to a core melt scenario. The component fragilities are used as input to evaluate the conditional fraction of failure of different safety systems as defined by the systems logic model.

Risk Quantification - The quantification of the likelihood of plant damage is accomplished by properly weighting the conditional fraction of plant damage (i.e., fragility) over the entire range of flood intensities, by their frequency of occurrence. By propagating the uncertainty in the frequency of flood hazards and fragility, the probability distribution on the likelihood of plant damage is evaluated.

3.4.2 External Flood Vulnerability

Site Description: ANO-1 is located on a peninsula on the east bank of Dardanelle Reservoir on the Arkansas River, approximately six miles west-northwest of Russellville, Arkansas. The plant site is at navigation mile 212 (river mile 264-3), located upstream of Dardanelle Lock and Dam (L&D) No. 10 which maintains a normal reservoir elevation of 338 feet National Geodetic Vertical Datum (NGVD). Approximately 50 miles upstream of ANO-1 is Ozark L&D No. 12. Plant grade at the site is 353 feet NGVD. The ground floor elevation of plant structures is 354 feet NGVD.

The topographic relief of the area surrounding the plant is sloping gently from the hills (about 500 feet NGVD) toward two major coves on the west bank of the Dardanelle Reservoir and a major cove of the Illinois Bayou on the east bank.

The principal cause of flooding in the area is the Arkansas River and its tributaries. Historically, the worst floods have occurred in the spring. In addition, intense rainfall can also cause local flooding. The largest recorded flood on the Arkansas River near ANO-1 occurred in May 1943. The peak discharge at Dardanelle, Arkansas (2 miles downstream from the dam) was 683,000 cubic feet per second (cfs). The flood reached a peak stage of approximately 343.5 feet NGVD which was 9.5 feet lower than plant grade of 353 feet NGVD. Dardanelle Dam is designed to discharge up to 900,000 cfs without exceeding a maximum water level of 338 feet NGVD.

The Arkansas River Navigation System consists of 17 locks and dams. Thirteen of these are simple locks and dams providing navigation lifts of 30 feet or less. Dardanelle, Ozark, Robert S. Kerr, and Webber Falls Dams are higher, with lifts up to 54 feet and include some storage for hydropower generation. Only Ozark Dam presents a potential threat to ANO-1 coincident with a PMF.

The original design, plant structures and components have been constructed to withstand the effect of extreme flood scenarios including local intense precipitation, upstream dam failure, and the probable maximum flood (PMF).

The flood level produced at the plant site due to the PMF is 358 feet. In the event that Ozark Dam fails during the PMF, a maximum water surface elevation of 361 feet, 8 feet above plant grade would result. The effect of wind generated waves was also considered in the plant design and it was concluded that wind generated waves do not constitute a significant hazard to the plant. The design basis flood elevation was therefore defined to be 361 feet.

Equipment Location: Safety-related equipment which is part of the shutdown decay heat removal system is located in the Reactor Building, Auxiliary Building, Turbine Building, and Intake Structure. Outside are the Condensate Storage Tanks (CST), Borated Water Storage Tanks (BWST), and Emergency Diesel Fuel Oil Storage Tank.

Category I plant structures (Reactor Building, Auxiliary Building, portion of Intake Structure housing, Emergency Diesel Fuel Storage Vault) have been evaluated to assess the effects of flood level elevation to plant grade. Safety-related equipment is either located on floors at or above elevations 369, or in Category I structures with wall thicknesses of at least 2 feet. Watertight doors are provided at entrances to the Auxiliary Building such that flood protection is provided to at least elevation 361 feet.

Based on a review of plant documents and a walkdown of the plant, an assessment of the location and vulnerability of safety-related equipment was made. Table 3.3 summarizes this information.

The switchyard is situated east of the plant, approximately at elevation 354 feet. Thus for floods that could impact the plant, failure of the switchyard and loss of offsite power is anticipated.

Emergency Procedures: In the event that a major flood occurs, emergency procedures exist to shut down the plant and to secure vulnerable areas. The procedures call for establishing contact with the Corps of Engineers to obtain accurate peak flood forecasts and updates should the lake level exceed elevation 340 feet.

According to the FSAR a flood of a magnitude of the PMF will be forecast about five days prior to arrival at the plant site. The plant is expected to be shut down by the time the flood level reaches 354 feet.

General Flood Considerations: In some cases special problems can arise due to local intense rainfall. These could include ponding on site, leakage into plant structures, and collection of water, snow, and ice on building roofs. As part of the screening analysis these areas were investigated. However, the results indicated that no further analysis is necessary as this was not a significant threat.

With respect to ice flooding, the FSAR indicates ice formation is light and does not cause significant flooding problems. On this basis it was concluded that a detailed hazard analysis for ice flooding is not required. Furthermore, according to the FSAR, wave action could accompany a large flood on the Arkansas River. Waves that occur on site should have little impact on

Table 3.3

VULNERABLE STRUCTURES AND EQUIPMENT

Structures/Equipment	Location	Vulnerability
480V Switchgear	Located on the east side of the Auxiliary Building, elevation 372 feet NGVD.	The 480V switchgear are vulnerable if flood elevations exceed 372 feet NGVD.
Emergency Switchgear Room	Located on the east side of the Auxiliary Building, elevation 372 feet NGVD.	The 4.16kV switchgear are vulnerable if flood elevations exceed 372 feet NGVD.
Emergency Feedwater Pumps	Located on the east side of the Auxiliary Building, elevation 335 feet NGVD.	Emergency Feedwater Pumps are vulnerable if the Auxiliary Building is flooded. Flood protection is provided to elevation 361 feet NGVD. Vulnerable if the exterior walls of the Auxiliary Building fail.
High Pressure Injection System (HPIS)	HPIS pumps are located on the north side of the Auxiliary Building, elevation 335 feet NGVD.	HPIS pumps are vulnerable if the Auxiliary Building is flooded. Flood protection is provided to elevation 361 feet NGVD. Vulnerable if the exterior walls of the Auxiliary Building fail.
Low Pressure Injection System (LPIS)	LPIS pumps are located on the northwest side of the Auxiliary Building, elevation 317 feet NGVD.	LPIS pumps are vulnerable if the Auxiliary Building is flooded. Flood protection is provided to elevation 361 feet NGVD. Vulnerable if the exterior walls of the Auxiliary Building fail.
Diesel Generators	Located on the northwest side of the Auxiliary Building, elevation 369 feet NGVD.	Vulnerable if flood elevation is above 369 feet NGVD.

Table 3.3

VULNERABLE STRUCTURES AND EQUIPMENT (CONTINUED)

Structures/Equipment	Location	Vulnerability
Battery Room	Located on the north side of the Auxiliary Building, elevation 372 feet NGVD.	Vulnerable if flood elevation is above 372 feet NGVD.
Cable Spreading Room	Located on the northeast side of the Auxiliary Building, elevation 374 feet NGVD.	Vulnerable if flood causes structural failure or leakage.
Control Room	Located on the north side of the Auxiliary Building, elevation 386 feet NGVD.	Vulnerable if flood causes structural failure or leakage.
Reactor Building Spray System (RBSS)	RBSS pumps are on the northwest side of the Auxiliary Building, elevation 317 feet NGVD.	Vulnerable if Auxiliary Building is flooded. Flood protection provided to elevation 361 feet NGVD. Vulnerable if the exterior walls of the Auxiliary Building fail.
Service Water Pumps	Located in the Intake Structure, elevation 366 feet NGVD.	Flood protection is provided to 361 feet NGVD.
Switchyard	Located east of the Turbine Building, elevation 354 feet NGVD.	Vulnerable if flood level exceeds elevation 354 feet NGVD.

plant structures. The principal effects of wave action would be spray and runup on the walls of reinforced concrete structures. These effects are not considered a significant hazard.

System Success Criteria: The assumption is made that a simultaneous loss of offsite power will occur during an external flood event since the switchyard is lower than the rest of the safety systems. The critical DHR functions are secondary cooling (using the EFWS) and emergency coolant injection in the feed and bleed mode (using the HPIS). Both the EFWS pumps and the HPIS pumps are at elevation 361'. The diesel generators and battery rooms are at higher elevations.

Risk Quantification: A flood of 361' fails the critical safety systems. Recovery of offsite power is not an issue since the critical safety pumps are submerged. The possibility of any other recovery actions is highly unlikely.

No credit is taken for operator actions even though emergency procedures have been established to shut the plant down should the Arkansas River exceed elevation 350'. This is because it is not known how long it would take to reach 361' NGVD from 350'. In addition, even though the plant was shut down, there still would be some residual heat that would need to be removed and no safety systems would be available to remove the heat. Thus the potential core melt probability is the $7.2E-6$ /year frequency of a 361' flood.

3.5 Extreme Wind Analysis

As part of the Task Action Plan A-45 analysis of the response of decay heat removal systems to internal and external initiating events, a study was performed for wind hazards for the ANO-1 Nuclear Power Plant. Both straight winds of excessive speed and tornado winds with their associated pressure differentials pose potential challenges to specific plant components. In addition, high winds can generate missiles with sufficient impact to penetrate doors, louvers, and metal buildings and severely damage water storage tanks, diesel generators, pumps, and other safety related systems.

These combined and potentially simultaneous threats of high wind force and wind generated missiles will be quantified briefly in this section. A more detailed description of the methodology, hazard analysis, component fragilities, and quantification will be presented in Appendix G of this report.

3.5.1 Analysis Methods

To evaluate the probability of plant damage due to external wind events, the following tasks are performed: 1) plant familiarization, 2) tornado and straight wind hazard analysis,

3) tornado missile and wind pressure fragility analyses, 4) systems analysis, and 5) risk quantification. Figure 3.4-1 shows the interrelationships of these elements.

Plant Familiarization - The purpose of this task is to gather information related to the occurrence of external wind hazards at a plant and the vulnerability of plant structures and equipment to wind effects. In assessing the reliability of decay heat removal systems, a review of available documentation (i.e., FSAR, plant drawings and topical reports) and a walkdown of the plant are performed. Potential missile population at the site, missile pathways, tank locations, equipment exposure, and building design are all visually evaluated for use in later analyses. Previous studies of structural design capacities and regional high wind frequencies are obtained as well as emergency operating procedures.

Hazard Analysis - Mean wind hazard curves for tornadoes and straight winds were developed by the Meteorology and Effluent Treatment Branch of the USNRC for use in the TAP A-45 program. The purpose of the hazard analysis is to develop the uncertainty in the hazard curves. Independent tornado hazard analyses are conducted which include consideration of the uncertainty in the tornado plant strike model, site size, tornado analysis data, tornado damage area and length variation, and Fujita F-scale intensity/wind speed conversion. The resulting curves are scaled so that the resulting mean frequencies of exceedance equal the USNRC values. An analysis of straight winds is conducted to develop the uncertainty in a similar manner as was done for tornados. Uncertainty in the wind speed data bases, applicability of recording station to the site, and terrain roughness differences are considered.

Tornado Missile and Wind Pressure Fragility Analyses - A tornado missile analysis was performed to develop the conditional probability distribution of missile impact on exposed equipment and on structures containing equipment which could fail due to missile penetration or by secondary impact from spalled concrete. The analysis considers the occurrence frequency of tornados, number of available tornado missiles, target exposure, and target area. Results of past tornado missile impact studies which were based on detailed missile simulations of mass, terminal velocity, deformation characteristics, and angle of impact are used. For structures and equipment vulnerable to wind pressure hazards, the cumulative fraction of failure defined as a function of the windspeed, called a fragility curve, is assessed.

Due to uncertainty, multiple fragility curves are determined for each component.

Systems Analysis - The decay heat removal systems analysis is performed by utilizing the event trees and fault trees

developed in the internal analysis to determine which systems are needed to prevent core melt. An event tree is then developed in which wind generated failures are an initiating event with a simultaneous loss of offsite power due to the severity of the storm. The component fragilities and missile impact factors are used to evaluate the conditional probability of failure of different safety systems.

Risk Quantification - The quantification of the likelihood of plant damage is assessed by properly weighting the conditional fraction of plant damage (i.e., fragility) over the entire range of wind intensities by the frequency of occurrence of wind hazards. Recovery actions for each sequence are given credit for where appropriate and, by propagating the uncertainty in the frequency of wind hazards and fragility, the probability distribution of the likelihood of plant damage can be evaluated.

3.5.2 Extreme Wind Vulnerability

ANO-1 is located on a peninsula of the Dardanelle Reservoir, near Russellville, Arkansas and is in an area with a relatively high rate of tornado occurrence. The site is also subject to straight wind hazards.

Vulnerable Structures and Equipment: In general, most safety-related equipment is contained within and protected by concrete structures which are at least 12 inches thick, and in many cases the barriers appear to be thicker. Equipment which is not protected and which is vulnerable to wind effects is listed in Table 3.4. It was assumed in the development of the vulnerable structure and equipment list that concrete walls and roof slabs greater than 12 inches thick provide adequate protection against tornado missiles. Doors, blockouts, exhaust stacks and other openings to the Category I structures were postulated to be potential vulnerable entrances to the buildings for tornado missiles. In general, all openings were either found to be protected by barrier walls, tornado missile barriers or too small to be significant contributors to the frequency of core melt. Based on a scoping analysis, openings less than 100 ft² correspond conservatively to a mean frequency of tornado missile impact less than 5.0E-7 per year; thus, the door openings were not found to be significant contributors and were not considered further.

The BWST, the CSTs, and the diesel oil storage tank are located outside and are vulnerable to wind effects. The tanks were not analyzed in this study for wind pressure effects based on experience with similar tanks at other nuclear power plants. Because the newly built 321,000 gallon tank was designed for seismic loads, the strength of this tank is sufficient to resist wind velocities greater than 360 mph. The other CST is not seismically designed but probably has a high strength

Table 3.4

VULNERABLE STRUCTURES AND EQUIPMENT

Structures/Equipment	Location	Vulnerability
Condensate Storage Tank (CST)	Southwest of Containment Building	Vulnerable to tornado missiles
Borated Water Storage Tank (BWST)	West of Auxiliary Building	Vulnerable to tornado missiles
Diesel Fuel Oil Storage Tank	Northeast of Cooling Tower	Vulnerable to tornado missiles
New 321,000 Gal. Storage Tank (CST)	West of Auxiliary Building	Vulnerable to tornado missiles (50,000 gallons of water are protected from missile impact by 5' high concrete wall).
Diesel Generator Exhaust Stacks	Auxiliary Building	Vulnerable to tornado missiles and wind pressure effects

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against wind pressure. However, these tanks are both vulnerable to tornado missiles and were included in the tornado missile analysis.

The exhaust stacks which go from the diesel generators up through the auxiliary building to the roof are exposed to tornado missiles above the roof. As indicated in the FSAR, the auxiliary building is designed to resist winds up to 300 mph, however the part of the auxiliary building where the exhaust stacks are attached consists of metal siding which is assumed to be Class 2 construction.

The safety-related buildings and structures at ANO-1 have been designed to resist wind pressures at least equal to 300 mph. This windspeed corresponds to a mean frequency of occurrence of approximately 2.0-6 per year. Because of conservatism in the design, the capacity is likely to be significantly higher and the corresponding mean frequency even lower. Based on this result the capacities of the buildings were not considered further. Thus, no detailed wind pressure analysis was required for the Class 1 part of the ANO-1 Nuclear Plant.

Wind Hazard: The ANO-1 site is exposed to potential winds generated by tornadoes and extratropical cyclones (including winter storms and summer thunderstorms), which are referred to in this report as straight winds. Because the site is located in the central portion of the U.S., there is no threat from hurricane winds.

A listing of all tornadoes within approximately a 145-mile radius of ANO-1 was obtained from the National Severe Storms Forecast Center (NSSFC) in Kansas City, Missouri. From 1950 through 1983 a total of 965 tornadoes were recorded of which 23 were rated Fujita scale F-4 (i.e., 207 to 260 mph) and one was rated F-5 (261 to 318 mph).

Straight wind data were obtained from the weather station at Little Rock, Arkansas which is approximately 57 miles southeast from the site. For the period 1952 to 1977 the largest recorded fastest mile windspeed is reported to be 61 mph, which occurred in 1974. ANO-1 is located in Region I of the USNRC tornado risk regionalization scheme given in WASH-1300. This region has the highest tornado hazard of the three USNRC regions. The mean values of wind speed frequencies are as follows:

<u>Tornado</u>		<u>Straight Wind</u>	
Windspeed (mph)	Mean Value	Windspeed (mph)	Mean Value
66	1.53E-3	79	1.00E-2
93	8.33E-4	84	5.00E-3
134	2.81E-4	91	2.00E-3
140	2.53E-4	97	1.00E-3
182	5.61E-5	102	5.00E-4
234	1.03E-5	109	2.00E-4
290	2.00E-6	115	1.00E-4
349	2.07E-7	128	2.00E-5
		133	1.00E-5
		140	5.69E-6
		146	2.00E-6
		151	1.00E-6

Sequence Quantification: The extreme wind core melt probability is the sum of the contributions from straight wind and tornado wind with missiles.

STRAIGHT WINDS - Only the diesel generators are vulnerable to potential straight winds at ANO-1. The failure probability of both diesel generators (DGs) is unity at 140 mph. This requires that both DG exhaust stacks fail. The probability that both stacks fail is simply the product of their two failure probabilities. In this case each failure probability is unity.

The core melt probability due to straight winds without recovery is 5.69E-6. This core melt probability is due solely to the failure of the DGs at 140 mph. If the battery operated EFW turbine driven pump starts and runs successfully, this is a long-term sequence since steam generator dryout will not begin until after battery depletion occurs in approximately 8 hours. It will take an additional 4 hours until steam generator dryout occurs. This is longer than the 30 minutes needed for a transient without the EFW system at the onset of the accident due to the reduced heat level after 8 hours. The nonrecovery probability of offsite power $p(\text{LOSP}_{\text{NR}})$ is 0.03 in 12 hours. In addition to LOSP recovery there is a possibility of recovery by having someone cut off the bent portions of the DG exhaust stacks (caused by the straight winds) within the 12 hours. This is given a nonrecovery probability of 0.1.

If the EFW turbine driven pump randomly fails to start, core melt will occur after approximately 30 minutes. The probability of nonrecovery of offsite power in 30 minutes is 0.5 and it is not considered possible that the DG exhaust stacks could be fixed in this time. Failure of the turbine driven EFW pump to start on demand is estimated to be $3.5E-2/\text{demand}$. Test and maintenance outage is a negligible contribution to EFW turbine pump failure. The incorporation of recovery is shown below for the loss of the DGs when the turbine driven pump successfully operates and when the turbine driven pump has failed to start.

$$p(\text{CM}|\text{STRAIGHT WIND}) = p(\text{DGs fail for 140 mph winds}) * [p(\text{LOSP nonrecovery in 12 hours}) * p(\text{nonrecovery of DG exhaust stacks by cutting open when bent by winds}) + p(\text{LOSP nonrecovery in 30 minutes}) * p(\text{TDP fails to start}) * p(\text{nonrecovery of DG exhaust stacks within 30 minutes})] * p(140 \text{ mph winds})$$

$$p(\text{CM}|\text{SW}) = 1.0 * [0.03 * 0.1 + 0.5 * 3.5E-2 * 1.0] * 5.69E-6$$

$$= 2.05E-2 * 5.69E-6 = 1.16E-7$$

Tornado Winds and Missiles: The core melt probability due to tornado winds and missiles is $2.53E-4$ if no recovery actions are considered. This core melt probability is due to failure of the DGs by tornado winds at 140 mph. It should be noticed that failure of the diesel fuel oil tank due to tornado missiles is not included since at ANO-1 there are two underground storage tanks plus two diesel day tanks. There is at least a seven day total diesel oil inventory available onsite in the emergency storage tanks for diesel generator operation during complete loss of electric power conditions.

The case where just the DGs fail is a long term sequence if the turbine driven EFW pump successfully runs, since steam generator dryout will not begin until after battery depletion occurs in approximately 8 hours, at which time the steam driven EFWS is assumed to fail. It will take an additional 4 hours until steam generator dryout occurs. This is longer than the 30 minutes needed for a transient without the EFW system at the onset of the accident. The nonrecovery of offsite power $p(\text{LOSP}_{\text{NR}})$ is 0.03 in 12 hours. In addition to LOSP recovery, there is a possibility of recovery in this case by having someone cut off the bent portions of the DG exhaust stacks (caused by tornado winds) within 12 hours. This is given a nonrecovery probability of 0.1.

If the EFW turbine driven pump randomly fails to start, core melt will occur after approximately 30 minutes. The probability of nonrecovery of offsite power in 30 minutes is 0.5 and it is not considered possible that the DG exhaust stacks could be fixed in this time. Failure of the turbine driven EFW pump to start on demand is estimated to be $3.5E-2/\text{demand}$. Test and

maintenance outage is a negligible contributor. The equation is the same as that used for straight winds with only the core melt probability without recovery differing.

$$p(\text{CM}|\text{Tornado Winds \& Missiles}) = 2.05\text{E-}2 * 2.53\text{E-}4$$

$$p(\text{CM}|\text{TW\&M}) = 5.19\text{E-}6$$

The contribution to the probability of core melt due to tornado winds and missiles is $2.53\text{E-}4$ without recovery and $5.19\text{E-}6$ with recovery. The total contribution to core melt due to straight winds and tornadoes (wind and missiles) is $2.59\text{E-}4$ without recovery and $5.31\text{E-}6$ with recovery.

3.5.3 Potential Modifications

The only wind vulnerability contributing greater than $1.0\text{E-}6/\text{R-yr}$ to the probability of core melt appears to be the collapse of the diesel generator exhaust stacks due to wind loading. The exhaust stack attachment to the building is vulnerable so that additional support is recommended. Construction of a wind barrier, provision for sturdier supports, or a design that ensures that the stack breaks free yet allows air flow would all reduce the vulnerability. Given such a modification the extreme wind core melt probability would become negligible.

3.6 Lightning Analysis

Each year at nuclear power plants lightning strikes occur which result in reactor trips, loss of offsite power, and equipment failures. The uncommonly high frequency of this special emergency requires an analysis of the consequences and potential contributions to core melt probability.

This section will present an overview of the methodology for evaluating the vulnerability of a power plant to lightning strike as well as quantification of the resulting probability of core damage for the ANO-1 Nuclear Power Plant.

Further details of analysis methods as well as a compilation of relevant Licensee Event Reports are presented in Appendix H of this report.

3.6.1 Analysis Methods

The evaluation of plant vulnerability to lightning strikes and subsequent determination of potential contribution to core melt probability performed in the TAP A-45 program utilizes the following steps:

- a. calculation of the lightning frequency of the site.
- b. calculation of the percentage of "damaging" strikes.

- c. evaluation of plant lightning protection.
- d. evaluation of critical components.
- e. evaluation of core melt probability contribution.

The calculation of lightning frequency is determined partially by the number of thunderstorm-days each year for a region and then by the number of cloud-to-ground strikes for a particular site latitude. The number of thunderstorm-days per year will be taken from a figure provided by the U.S. Meteorological Service and included in Appendix H. Several correlations exist to calculate the number of cloud-to-ground strikes for various regions of the United States. For this study, the relationships to be used are:

<u>Location</u>	<u>Ground Flash Density ($\text{km}^{-2}\text{yr}^{-1}$)</u>
Northern U.S.	0.11T
Southern U.S.	0.17T

where T is the number of thunderstorm-days for the locality of interest. To confirm this correlation and to obtain the range of possible values, a second correlation developed by Pierce will be used as follows:

$$\text{Cloud-to-ground flash density} = (0.1 + 0.35 \sin \text{latitude}) \times (0.40 \pm 0.20)T \text{ km}^{-2}\text{yr}^{-1}.$$

The percentage of these ground strikes which can damage a plant are highly plant specific and dependent upon the adequacy of lightning protection systems. NRC recommendations suggest that plants be designed to withstand a strike of up to 200 kilo-amperes (kA). Since it is beyond the scope of this program to verify the successful operation of these devices, it will be assumed that existing plant lightning protection is adequately designed and maintained. Studies indicate that between one out of one hundred and one out of every thousand ground strikes exceeds 200 kA in current. In this program, we will estimate the probability of a lightning strike being powerful enough to exceed the protective features to be .01. That is, one strike out of every hundred will have a current greater than 200kA and will be able to damage the plant.

The evaluation of plant protection will be extended only to a literature review, a site visual confirmation, and discussions with plant personnel. The literature review will be limited to studies performed by the plant or information provided in the Final Safety Analysis Report which describe the lightning protection and ground grid system. A site visit is then conducted where the air terminals, earth shield wires and ring conductors are visually confirmed. Additional discussions are held with plant personnel to fill any informational gaps concerning the ground grid system or past lightning occurrences.

The evaluation of critical components is performed in conjunction with the internal analysis of transient initiated events. This approach allows the analyst to identify the success criteria of front line and support systems needed to assure decay heat removal following a reactor trip, loss of offsite power, or other transient which could be caused by a lightning strike. If multiple systems, or multiple trains of a single system, are available for decay heat removal, their sources of water, electric power, instrumentation and control, etc., must be investigated for common mode failures. Again, use of the internal analysis fault tree methodology allows a straight forward approach to this analysis. This approach results in a concentrated list of front line systems and necessary support systems which will prevent core melt if protected against either simultaneous lightning initiated failures, or a lightning strike in conjunction with any number of random system failures.

The calculation of the probability of failure of decay heat removal systems following a lightning strike is the final step in quantifying the contribution to core melt. Given the frequency of lightning striking the site with a current greater than 200kA, estimates must be made concerning the probability of striking a critical system. If a single system must be hit to lead to core melt, the assumed probability of such a critical hit, given all of the available targets on the plant site, is very conservatively assumed to be one in ten (0.1). If a single component, electrical bus, battery, etc., is the necessary target, the probability of a damaging strike, given all of the available targets, is assumed to be one in one hundred (0.01). However, if a second independent train is available for successful operation, then the probability of either a second lightning strike or a random system failure must be multiplied by the initiating system failure probability. Multiple systems would likewise require multiple lightning strikes or multiple random failures according to the following equation:

$$\begin{aligned} \text{Core Melt Probability} = & T \left[\left(\frac{Nq}{T} \right) (A) (P_{kA}) (P_{hit}) \right]^n (LOSP) (NR) + \\ & (R_1) (T) \left[\left(\frac{Nq}{T} \right) (A) (P_{kA}) (P_{hit}) \right]^{n-1} (NR) + \dots + \\ & (R_1) (R_2) \dots (R_{n-2}) (R_{n-1}) T \left[\left(\frac{Nq}{T} \right) (A) (P_{kA}) (P_{hit}) \right] (LOSP) (NR) \end{aligned}$$

where T = Number of thunderstorm-days per year for the region

Nq = Number of lightning strikes per km^2 -year

A = Area of the plant site in km^2

P_{kA} = Probability that the lightning current exceeds the design basis

- P_{hit} = Probability that lightning hits a critical component
 I_{OSP} = Probability of loss of offsite power
 NR = Probability of nonrecovery (0.1)
 n = Number of independent, redundant trains available to prevent core melt
 R_n = Probability of random failure of the n_{th} train

3.6.2 Lightning Vulnerability

Using the U.S. Meteorological Services chart of thunderstorm-days in the United States (see Appendix H) for ANO-1, it is estimated that the site would expect approximately 58 thunderstorm-days each year (T). The correlations for ground flashes (N_g) for the southern U.S. derived by Horn and Ramsey is $N_g = 0.17(T)$. This correlation predicts 9.9 ground flashes $\text{km}^{-2}\text{yr}^{-1}$ for this southern U.S. site. ANO-1 is located at latitude 35.5°N which, using the equation derived by Pierce, predicts a ground flash density range of:

$$N_g = (0.1 + 0.35 \sin \text{latitude})(0.40 \pm 0.20)T \text{ km}^{-2}\text{yr}^{-1}$$

$$\text{or } N_g = 3.5 \rightarrow 10.4/\text{km}^2\text{yr}.$$

This is one estimate of the range of ground flash densities for ANO-1 while the value of $N_g = 9.9$ will be used as a point estimate of the number of strikes per km^2 per yr even though it is near the upper bound of the range estimate.

It is assumed from the literature review, site inspection, and discussions with plant personnel that the plant was designed according to NRC recommendations that the plant be able to withstand a lightning strike of 200kA. The probability of exceedance of this current is taken to be 0.01 as seen in Appendix H, Figure H-2. Therefore, the probability of lightning strikes in the ANO-1 region which will exceed the design basis is estimated to be $(0.01) \times (9.9) = 0.099/\text{km}^2\text{yr}$. To simplify calculations, the ANO-1 site will be conservatively estimated to be $1/4$ square kilometer in size.

The transient event trees developed for ANO-1 in this program show that either the emergency feedwater system or the feed and bleed mode will successfully remove decay heat following a transient. The emergency feedwater system consists of two trains such that the probability of random failure of both trains is small and a single lightning strike should not fail both of the independent trains. However, electric power is needed for the starting and running of pumps, operation of valves, and instrumentation and control. Therefore, the

limiting core melt scenario would require loss of both DC electric power trains.

Each DC bus can receive power from offsite power, the associated diesel generator, or the batteries dedicated to that train; however, the diesel generator requires the DC batteries in order to start. The scenario where both DC power trains fail due to random or common mode failures is part of the internal analysis of Chapter 2. In this section, the postulated transient involves either two lightning strikes hitting the two DC buses, or a single lightning strike hitting one DC bus while the other DC power train randomly fails. Core melt will commence in less than 30 minutes when the steam generators have boiled dry.

These sequence probabilities are calculated as follows:

$$\text{Core Melt Probability} = T \left[\left(\frac{N_g}{T} \right) (A) (P_{kA}) (P_{hit}) \right]^2 (NR) +$$

$$T \left[\left(\frac{N_g}{T} \right) (A) (P_{kA}) (P_{hit}) \right] (LOSP) (NR) (R_1)$$

where T = 58 thunderstorm-days/yr
 N_g = (9.9 ground flashes/km²yr)
 A = 0.25 km²
 P_{kA} = 0.01
 P_{hit} = 0.1
 $LOSP$ = 0.5
 $NR_{30 \text{ min}}$ = 0.5

It should be noted that the term denoted P_{hit} is a conservative assumption that one out of one hundred lightning strikes greater than 200kA will damage this particular DC bus. This assumption is probably too high, however, it is beyond the scope of this analysis to trace the pathways of lightning pulses and then assign probabilities to these pathways. The use of a value of 0.1 for P_{hit} will yield an estimate for the damage of a DC bus.

Random failure of a DC power train following a loss of offsite power would require either failure of the bus itself or loss of the power supply from the battery. The failure of the battery will cause failure of the emergency diesel generator to start.

The calculation for random failure of a DC train of power, then, is as follows:

$$R_1 = (\text{DC bus local fault}) + (\text{DC bus test or maintenance}) + (\text{Battery local fault}) + (\text{Battery test or maintenance})$$

The following data is used in this evaluation:

$$P_f(\text{BATT})_{LF} = 1.8\text{E-}3 \text{ probability of failure of a battery due to local faults}$$

$$P_f(\text{BATT})_{UTM} = 1.1\text{E-}3 \text{ unavailability of a battery due to test or maintenance}$$

$$P_f(\text{DCBUS})_{LF} = 3\text{E-}5 \text{ local fault failure probability for a DC bus plus one circuit breaker}$$

$$P_f(\text{DCBUS})_{UTM} = \epsilon$$

Therefore:

$$R_1 = (3\text{E-}5) + (\epsilon + (1.8\text{E-}3) + (1.1\text{E-}3)) = 2.9\text{E-}3$$

The probability of core melt is then calculated to be:

$$p(\text{CM}|\text{lightning}) = 58 * \left[\frac{9.9}{58} * 0.25 * 0.01 * 0.01 \right]^2 * 0.5 * 0.5 +$$

$$58 * \left[\frac{9.9}{58} * 0.25 * 0.01 * 0.01 \right] * 0.5 * 0.5 * 0.0029$$

$$= 2.64\text{E-}10 + 1.79\text{E-}7 = 1.79\text{E-}7/\text{R-yr}$$

For the case where lightning strikes the 125V DC B train and the train A diesel generator fails to start, the A train of DC power will be available for instrumentation and the AFW steam supply valves to the turbine driven pump as well as one feedwater valve could be manually opened. The safety grade batteries are sufficient for eight hours of electric power without recharging.

3.6.3 Potential Modifications

It readily can be seen that all assumptions are extremely conservative in this simplified analysis. Nonetheless, even the worst case lightning scenario does not contribute significantly to the overall core melt probability. In addition, proposed internal analysis modifications for ANO-1 include a turbine driven generator which will lower the random failure probability even more.

It was impossible in this program to analyze the subtleties of lightning strikes damaging the control room and then assess the possible consequences. However, it is recommended that this area be adequately protected against all lightning strikes according to the Lightning Protection Code NFPA 78. The high

incidence of lightning strikes suggests that adequate protection is imperative.

3.7 Sabotage Analysis

Prior studies have suggested that unauthorized activities by insiders (i.e., plant personnel) can be a major source of vulnerabilities. Such acts may include deliberate valve misalignment, improper set point adjustments, physical damage to small diameter cooling lines (e.g., oil and bearing cooling on safety pumps and diesel generators), improper installation of bearings or couplings and so on. Many such acts, in and of themselves do not cause core melt or a significant radiological release, but given an initiating event such as a loss of offsite power, the failure of these safety components can lead to an increase in public risk. It should be noted that in these investigations the use of explosives by insiders was not considered. This was based upon the earlier studies which suggest that there is such a variety of ways an insider can disable equipment that explosives are not really required. Also, it is presumed that current security measures will prevent the introduction of commercial explosives into the plant.

For purposes of this program, direct overt acts on the part of an outside group are not considered. The evaluation of site security is outside the scope of these investigations. It is presumed here that those plants which meet the requirements of 10 CFR 73.55 are capable of adequately dealing with the external threat. Further, the emphasis on these insider activities is directed toward the potential system vulnerabilities and what can be done to reduce or eliminate them. The analysis takes note of existing administrative and security policy and procedures as appropriate, including their potential influence upon insider activities, but it does not examine in detail the effects of changes in such procedures.

Because of the potential sensitivity of detailed discussions of sabotage vulnerabilities, they are not summarized here but are discussed separately in Appendix I. That appendix is issued separately under appropriate SAFEGUARDS controls.

3.8 Summary of Special Emergency Modifications

Table 3.5 presents a compilation of the ANO-1 dominant vulnerabilities for each special emergency analysis performed in this study. The table lists the initiating event and then identifies the vulnerable area or specific component(s). The core melt probability is given both before and after any proposed modification. In the case of the seismic analysis, since the entire plant "sees" the ground acceleration, the core melt probability is not due to failure of any one system.

Sabotage vulnerabilities are now shown on Table 3.5, but, as noted in Section 3.7, they are discussed in Appendix I.

Table 3.5 Summary Table of Special Emergency Modifications

Initiating Event	Vulnerable Area/Component	Core Melt Probability/R Yr	Proposed Modification	Modified Core Melt Probability/R Yr
Seismic	Borated water storage tank & condensate storage tank		Add braces to the BWST and CST	
	Electrical cabinets		Provide additional anchorage for the 4160 V safeguards buses, 480 V transformer buses, and two Motor Control Centers in the switchgear room	
Seismic Total = $7.3E-5$			Seismic Fixes = $8.8E-6$	
Fire	Cable spreading room	$5.8E-6$	Install redundant deluge valve controlled by additional sensing system	$8.1E-7$
External Flood	Turbine and Auxiliary Bldgs.	$7.2E-6$	None	-----
Extreme Wind	DC exhaust stacks	$5.3E-6$	Provide additional structural steel strap supports for each exhaust line	c
Lightning	DC buses	$1.8E-7$	None	-----

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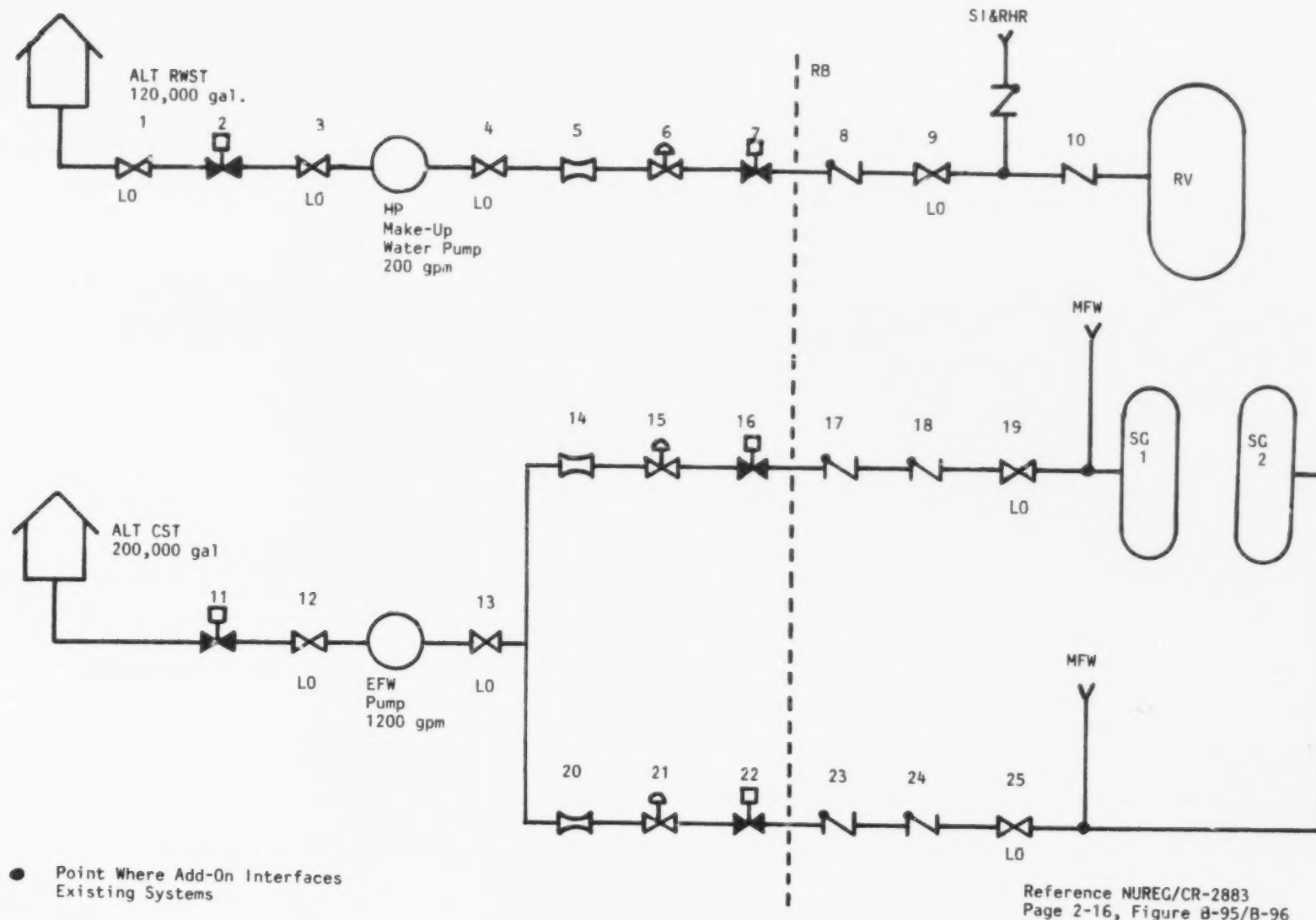
4.0 DEDICATED SHUTDOWN DECAY HEAT REMOVAL SYSTEM

The add-on dedicated shutdown decay heat removal (SDHR) system suggested for Point Beach is a single train of high pressure makeup water and a single train of emergency feedwater with dedicated power and water sources. This type system was selected based upon results from earlier NRC studies (Reference 11) and upon the belief that forced or natural circulation through the steam generators provides the most effective path for decay heat removal from the core. Such a system is also consistent with much of the current European practice (Reference 12).

The SDHR system is a backup system designed to provide emergency core cooling in the unlikely event of a failure of the existing safety systems during a SBLOCA or transient. Core cooling is accomplished by injection of emergency feedwater into the steam generators and release of steam via dedicated atmospheric dump valves. Natural circulation in the reactor coolant loops insures flow and heat removal in the core. The reactor coolant system pressure is maintained above saturation pressure by the use of one group of pressurizer heaters in conjunction with the alternate makeup system. Offsite power will be used when available, otherwise the add-on system is completely independent of the plant systems. The system consists of a high pressure emergency feedwater type pump, a high pressure makeup pump, a storage tank for borated water, a storage tank for feedwater, and the required piping, valving, instruments, and controls for initiation, monitoring and operation of the system. This system is automatically actuated given failure of the primary systems. The structures and connections to the existing system are designed to seismic Category I specifications. A complete description of this system can be found in Reference 11 and in the impact analysis Appendix J. The rationale for a single train system housed in Seismic I structures as contrasted with other options; two-trains, bunkered building, etc., is discussed in Section 6 and 7.

The probabilistic analysis developed here will provide the reliability information for this system required for the value analysis in Section 7. This probabilistic analysis uses the simplified piping diagram shown in Figure 4.1.

The fault tree (Figure 4.2) for the add-on system considers that both the high pressure make up subsystem and the emergency feedwater subsystem are necessary for total system success. Electric power (EP) which is typically attached to each component it supports is brought into the top gate since it is clear that EP is required for system success and the Boolean reduction of the fault tree model with EP on each component it supports would lead to the same result. Room cooling for the pumps is not required. Fans to cool the diesel generator are needed and powered off the diesel generator, but are not modeled explicitly.



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Figure 4.1 Simplified Schematic for Dedicated Add-On With
Emergency Feedwater and High Pressure Make-Up Water

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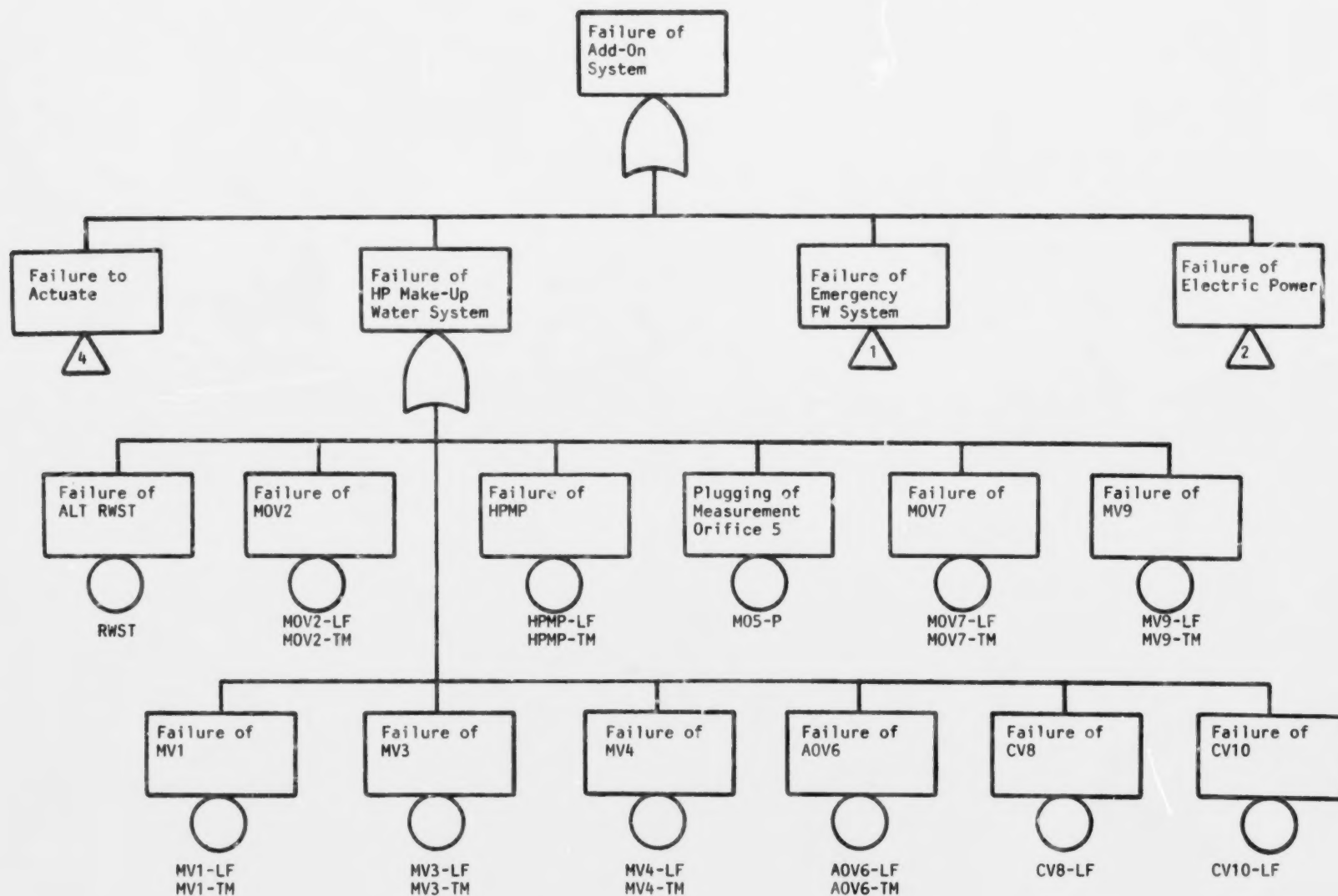


Figure 4.2 Potential Add-On SDHR System Fault Tree

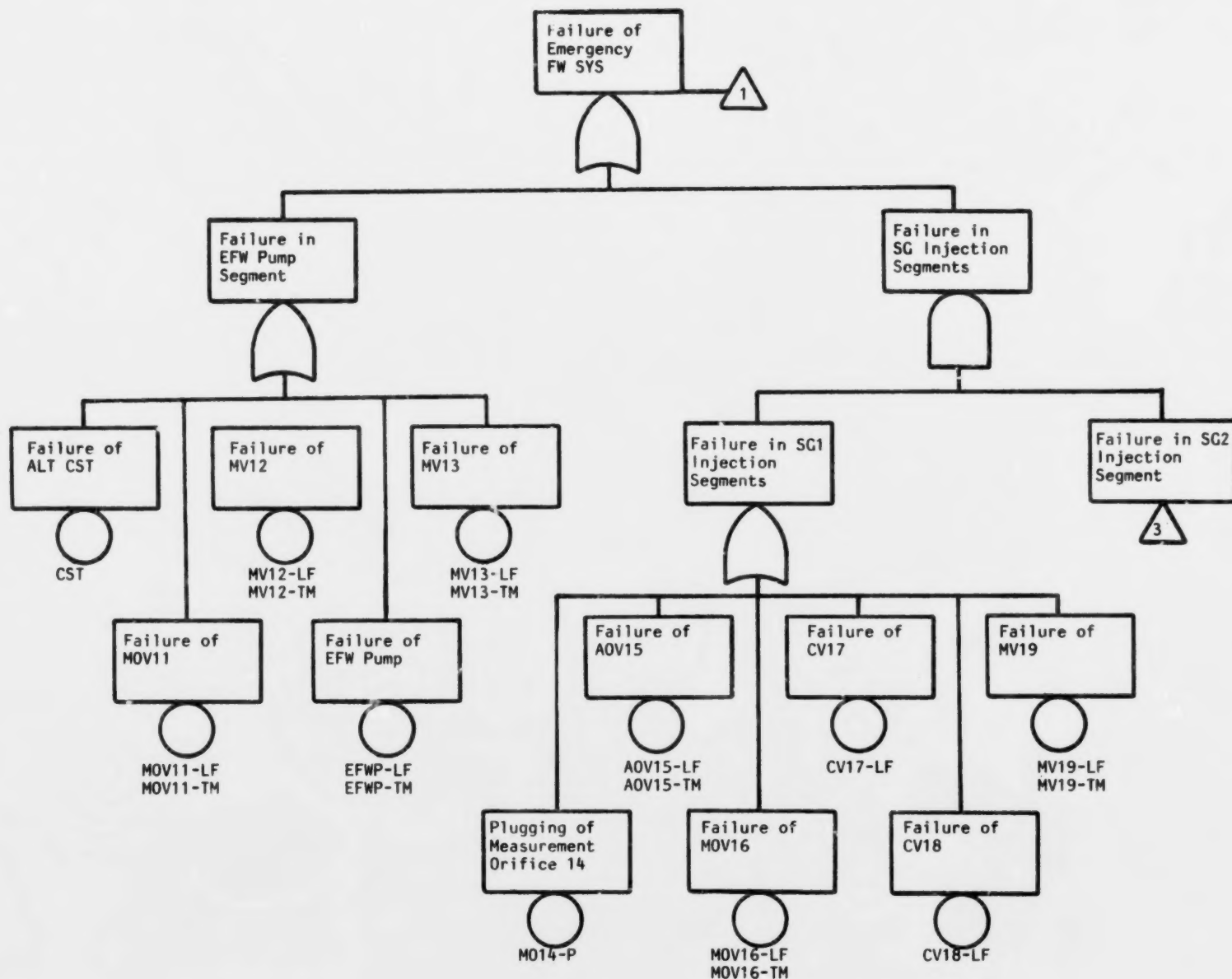


Figure 4.2 Potential Add-On SDHR System Fault Tree (cont.)

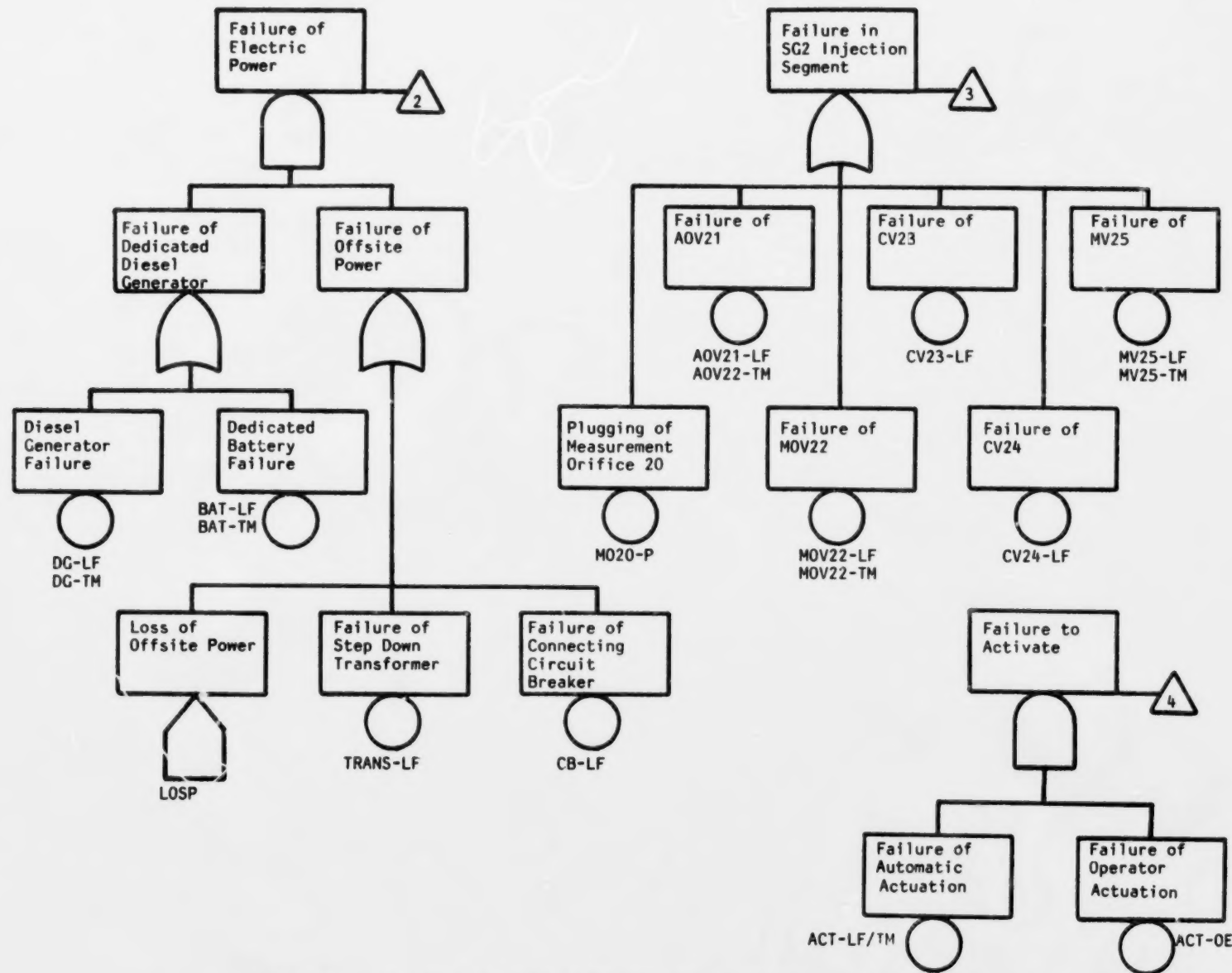


Figure 4.2 Potential Add-On SDHR System Fault Tree (cont.)

Each of the components shown in the system diagram (Figure 4.1) is depicted in the fault tree (Figure 4.2). If there are multiple failures for the component, these are simply shown below the event box. The acronyms used in the event names are:

Failures:	OE	Operator Error
	LF	Local Fault (hardware)
	TM	Test & Maintenance
	P	Plugging
Event Names:	ACT	Actuation
	RWST	Refueling Water Storage Tank
	CST	Condensate Storage Tank
	MV	Manual Valve
	MOV	Motor Operated Valve
	AOV	Air Operated Valve
	CV	Check Valve
	HPMP	High Pressure Makeup Pump
	EFWP	Emergency Feedwater Pump
	DG	Diesel Generator
	BAT	Battery
	TRANS	Transformer
	CB	Circuit Breaker
	LOSP	Loss of Offsite Power
	MO	Measurement Orifice

Several assumptions were made in the fault tree model, quantification, and subsequent use in the value analysis. These are:

- 1) All the equipment and the structure containing the equipment are designed to withstand all special emergencies.
- 2) All the equipment is designed to standard safety equipment specifications.
- 3) There are no common mode interactions between the redundant discharge legs of the emergency feedwater subsystem.
- 4) Rupture of the add-on system piping has an insignificant failure probability and thus is not modeled.
- 5) Offsite power directly from the switchyard will be used to power the add-on system whenever it is available.
- 6) The system will be automatically actuated with backup actuation by the reactor operator.

Since there are no dependent or common events between branches of the fault tree, the quantification can be accomplished directly from the fault tree without any complicated Boolean manipulation. The event failure probabilities are given in Table 4.1. The expression representing the fault tree is given below in equation form with each Boolean event and its

Table 4.1 Add-On System Value Block

<u>Event</u>	<u>Failure Probability</u>	<u>Source</u>
ACT-OE	3E-3	Ref. 8, Table 20-7, item 3, short list, no checkoff
ACT-LF	5E-3	Ref. 9, Table B10-2
RWST	0	No observed tank failures
CST	0	No observed tank failures
MOX-P	3E-4	Ref. 10, Table 5.1-1, item 4.4.1
HPMP-TM	2.1E-3	Ref. 8, Table 20-7, item 1, failure to restore after T&M (1E-3)(1/12 monthly check) plus 2E-3 due to downtime
EFWP-TM	2.1E-3	Ditto
HPMP-LF	9E-3	Ref. 10, Table 5.1-1, item 1.1 (3E-3/d) + (3E-5/h) x 24 hrs + (5E-3) control circuit failures
EFMP-LF	9E-3	Ditto
CVX-LF	1E-4	Ref. 10, Table 5.1-1, item 2.4.1
MVX-LF	1E-4	Ref. 10, Table 5.1-1, item 2.6.2 (1/2) x (3E-7/h)x(720 hrs) monthly test interval
AOVX-LF	4E-5	Ditto except 1E-7/h from item 2.1.2
MOVX-LF	8E-3	Ref. 10, Table 5.1-1, item 2.1.1, failure to open on demand (3E-3) plus control circuit failures (5E-3)
MVX-TM	3.8E-4	Ref. 8, Table 20-7, item 1, failure to restore after T&M (1E-3)(1/12 monthly check) plus (3E-4) due to downtime
AOVX-TM	3.8E-4	Ditto
MOVX-TM	3.8E-4	Ditto
DG-TM	6.1E-3	Ditto except 6E-3 due to downtime
BAT-TM	1.1E-3	Ditto except 1E-3 due to downtime
DG-LF	3.8E-2	See Appendix B3.3.4
BAT-LF	3.9E-3	See Appendix B3.3.1
TRANS-LF	2.4E-5	Ref. 10, Table 5.1-1, item 4.5.1 (1E-6/h) x 24 hrs
CB-LF	3E-5	Ref. 10, Table 5.1-1, item 4.1.2
LOSP	1	For LOSP initiated transients
	0	For all other initiating events since the failure frequency of LOSPs already included as 1E-3
MOV-CM	4E-4	Ref. Appendix B, Section 3.3.4

X $\equiv$ variable depending on the component number Table 4.1

associated probability; for example, ACT-OE/3E-3 means that 3E-3 is the probability of the event ACT-OE, actuation operator error.

Failure of the Add-on System = the sum of

$$\begin{aligned} & (\text{ACT-OE}/3\text{E}-3 * \text{ACT-LF}/5\text{E}-3) + \text{RWST}/0 + \text{MV1-LF}/1\text{E}-4 + \\ & \text{MV1-TM}/3.8\text{E}-4 + \text{MOV2-LF}/3\text{E}-3 + \text{MOV2-TM}/3.8\text{E}-4 + \text{MV3-LF}/1\text{E}-4 + \\ & \text{MV3-TM}/3.8\text{E}-4 + \text{HPMP-LF}/9\text{E}-3 + \text{HPMP-TM}/2.1\text{E}-3 + \text{MV4-LF}/1\text{E}-4 + \\ & \text{MV4-TM}/3.8\text{E}-4 + \text{MO5-P}/3\text{E}-4 + \text{AOV6-LF}/4\text{E}-5 + \text{AOV6-TM}/3.8\text{E}-4 + \\ & \text{MOV7-LF}/8\text{E}-3 + \text{MOV7-TM}/3.8\text{E}-4 + \text{CV8-LF}/1\text{E}-4 + \text{MV9-LF}/1\text{E}-4 + \\ & \text{MV9-TM}/3.8\text{E}-4 + \text{CV10-LF}/1\text{E}-4 + \text{CST}/0 + \text{MOV11-LF}/8\text{E}-3 + \\ & \text{MOV11-TM}/3.8\text{E}-4 + \text{MV12-LF}/1\text{E}-4 + \text{MV12-TM}/3.8\text{E}-4 + \text{EFWP-LF}/9\text{E}-3 + \\ & \text{EFWP-TM}/2.1\text{E}-3 + \text{MV13-LF}/1\text{E}-4 + \text{MV13-TM}/3.8\text{E}-4 + (\text{MO14-P}/3\text{E}-4 + \\ & \text{AOV15-LF}/4\text{E}-5 + \text{AOV15-TM}/3.8\text{E}-4 + \text{MOV16-LF}/8\text{E}-3 + \\ & \text{MOV16-TM}/3.8\text{E}-4 + \text{CV17-LF}/1\text{E}-4 + \text{CV18-LF}/1\text{E}-4 + \text{MV19-LF}/1\text{E}-4 + \\ & \text{MV19-TM}/3.8\text{E}-4) * (\text{MO20-P}/3\text{E}-4 + \text{AOV21-LF}/4\text{E}-5 + \\ & \text{AOV21-TM}/3.8\text{E}-4 + \text{MOV22-LF}/8\text{E}-3 + \text{MOV22-TM}/3.8\text{E}-4 + \\ & \text{CV23-LF}/1\text{E}-4 + \text{CV24-LF}/1\text{E}-4 + \text{MV25LF}/1\text{E}-4 + \text{MV25-TM}/3.8\text{E}-4) + \\ & (\text{DG-LF}/3.8\text{E}-2 + \text{DG-TM}/6.1\text{E}-3 + \text{BAT-LF}/3.9\text{E}-3 + \text{BAT-TM}/1.1\text{E}-3) * \\ & (\text{LOSP} + \text{TRANS-LF}/2.4\text{E}-5 + \text{CB-LF}/3\text{E}-5) + \text{MOV-CM} \end{aligned}$$

After the probabilities for the basic events are summed the equation reduces to three terms.

$$(5.12\text{E}-2) + (9.78\text{E}-3) * (9.78\text{E}-3) + (4.91\text{E}-2) * (\text{LOSP} + 5.4\text{E}-5) + 4\text{E}-4$$

When the terms are multiplied out and combined, the equation further reduces to:

$$(5.17\text{E}-2) + (4.91\text{E}-2) * \text{LOSP}$$

This indicates that the add-on system failure is dominated by seven events in the first term (i.e., MOV2-LF, HPMP-LF, HPMP-TM, MOV7-LF, MOV11-LF, EFWP-LF, and EFWP-TM) and LOSP with DG-LF, DG-TM, and BAT-LF. In other words, the probability of failure lies in the operating components and electric power. Finally, considering the two possible conditions for availability of electric power, the system failure probabilities become:

$$\begin{aligned} & p(\text{add-on system failure given offsite power is available}) \\ & = 5.2\text{E}-2 \text{ per demand} \end{aligned}$$

and

$$\begin{aligned} & p(\text{add-on system failure given offsite power is unavailable}) \\ & = 1.0\text{E}-1 \text{ per demand.} \end{aligned}$$

These values will be used in the value analysis outlined in Section 7.

5.0 ALTERNATIVE SELECTION AND INTEGRATION

Fifteen individual potential vulnerabilities were identified during the initial internal and special emergency core melt probability analyses for the ANO-1 Nuclear Plant. These vulnerabilities are given in Table 5.1 with the approximate core melt frequency attributed to each of them. Even though the vulnerabilities and associated modifications described here were identified in a probabilistic type analysis, the diversity of the modifications is such that many potential difficulties that might be envisioned are covered. Furthermore, as has been the experience in other programs, it is the support systems, not the front line systems which cause problems. Sabotage vulnerabilities are not explicitly shown in Table 5.1. A number of the special emergency vulnerabilities have sabotage implications so that it is implicitly included here. All of these vulnerabilities and the corresponding modifications are discussed in more detail in the appropriate subsections of Sections 2.0 and 3.0. Combinations of modifications will become the basis for the alternatives.

In the early stages of the program during the safety criteria screening process, a number of insights regarding potential vulnerabilities were identified. These insights are listed in Table 5.2. Some of these early insights correlate well with the vulnerabilities found in the subsequent analyses. However, many of the insights are out of date or not applicable to our analysis, e.g., the insight related to the subcriticality function. Each of the seven insights are briefly discussed below.

1. The single discharge valve from the BWST was found to be important and is addressed by the modification proposed for vulnerability number 8.
2. The transfer from the BWST to the SUMP, i.e., ECI to ECR, is done manually and is discussed in vulnerability number 7. While increased operator awareness and training is suggested, this particular action was not significant.
3. The motor driven EFW pump is now automatically initiated and is analyzed accordingly.
4. We did not address the issue of a remote shutdown in the internal analysis, however, reference is made in the special emergency analyses.
5. The single line header from the CST to the EFWS was not a significant core melt probability item.
6. ANO is constructing a new seismic category I CST.

Table 5.1 ANO-1 Potential Internal and External Vulnerabilities

<u>Vulnerabilities</u>		<u>Approx. Contribution to Core Melt Probability of Sequences Involving This Vulnerability</u>
Internal 1	EFWS Turbine Driven Pump Failure	7E-6
Internal 2	Common Mode Valve Failures	2E-5
Internal 3	Common Mode Pump Failures	3E-6
Internal 4	Diesel Generator Failures	5E-6
Internal 5	Battery Failures	2E-5
Internal 6	Low Pressure Pump Failures	1E-5
Internal 7	Failure to Manually Actuate	5E-6
Internal 8	BWST Discharge Valve Failure	6E-6
Seismic	Switchgear & Control Rooms, BWST & CST	7E-5
Fire	Cable Spreading Room	6E-6
Lightning	Fails Both DC Trains	2E-7
Extreme Wind	DG Exhaust Stacks	5E-6
Ext. Flood	River Flooding and Dam Failure	7E-6

Table 5.2 ANO-1 Insights from Study Performed for
TAP A-45 by ORNL in 1983

1. Single outlet valve from BWST.
2. Transfer from ECI to ECR in a manual operation.
3. Motor driven EFW pump manually initiated.
4. No remote shutdown panel.
5. Single line header from CST to EFWS.
6. CST is not seismic category 1.
7. Auxiliary Feedwater pump is not seismic category 1 or class 1E.

7. The Auxiliary Feedwater pump insight is addressed partially by the modification to vulnerability 7 which adds a direct line to the CST and makes the power class 1E.

While modifications were suggested for most of the vulnerabilities identified, only 10 modifications were carried into the impact analysis. Each of these modifications addresses a specific vulnerability and in a few cases more than one vulnerability. The dedicated SDHR system described in Section 4.0, however, applies to all of the vulnerabilities, internal and external, including sabotage. Combinations of modifications for specific vulnerabilities and/or the dedicated SDHR system are called alternatives. Two alternatives were selected as shown in Table 5.3.

As noted above, these vulnerabilities were identified on the basis of core melt probability, but their scope is broad enough to include nonquantified events also. The total core melt probability for those initiating events covered in this program is greater than the sum of the individual vulnerability contributions due to the residual core melt probability that is attributable to internal accident sequences and cut sets that were not dominant and to special emergency events that were not quantified. The base case core melt probability is discussed in detail in Section 7.1.1. The maximum potential value that can be obtained by a modification is the core melt probability of the vulnerabilities it addresses, i.e., to have a perfect modification which reduces that vulnerability to zero.

The alternatives in Table 5.3 were selected to obtain the potential benefits of plant modification vs the add-on DHR system. Table 5.4 gives the preliminary estimates used in establishing the alternatives. Alternative one was selected to address all of the external vulnerabilities and internal vulnerabilities. Alternative four addresses the add-on SDHR system alone. The impact analysis for these alternatives is discussed in Section 6 and the anticipated reductions in core melt probability based on the analysis of the conceptual designs are presented in Section 7.

Table 5.3 ANO-1 Potential Alternatives

<u>Modification:</u>	<u>Alternatives</u>	
	<u>1</u>	<u>2</u>
Internal 1	X	
4	X	
5	X	
6	X	
8	X	
Seismic 1-3	X	
Fire	X	
Wind	X	
 <u>Dedicated SDHR System</u>		
Single Train Add-on		X

Table 5.4 Preliminary Estimated Core Melt
Probabilities Given Modifications
Applied to ANO-1

Total Core Melt Probability Before Modifications	2.3E-4
Alternative 1	5.0E-5
Alternative 2	1.7E-5

6.0 ALTERNATIVE IMPACT ANALYSIS

The impact analysis addresses the two alternatives identified in Section 5.0 which consist of various combinations of sixteen potential plant modifications including a dedicated shutdown decay heat removal system. Each of the modifications is first considered separately, then combined according to the alternative definitions to account for interactions between modifications during construction in an alternative such as scheduling or accessibility conflicts.

6.1 Methodology and Approach

6.1.1 Objectives

The objectives of the impact assessment were to determine all the potential impacts resulting from the implementation of each alternative. These are:

- 1) Cost of equipment and installation of the modifications included in a specific alternative in 1985 dollars,
- 2) Cost of replacement power, if required, in order to implement an alternative in 1985 dollars,
- 3) Cost of subsequent annual operation, maintenance, and in-service inspections that would be associated with the alternative in 1985 dollars, and
- 4) Radiation exposure involved in the installation of the alternative in person-rem.

It was also desired to obtain potential yearly changes to occupational radiation exposure due to the alternatives and replacement power costs if the alternatives would affect the normal outage times. These two impacts could be positive or negative theoretically, however, they were both judged to be negligible for the ANO-1 Plant.

6.1.2 Approach

Given the vulnerability identification and proposed alternatives, initial design reports were prepared on each modification. These reports described the proposed design, interfaces with other systems and structures, the scope of work to be performed such as bill of materials and rip-out required, and major construction activities. A site inspection plan was prepared for each modification which identified all the specific areas to be visited and questions to be asked in order to properly estimate the impact of the alternatives. The preliminary design was done based upon drawings such as building layouts and piping and instrumentation drawings that were obtained prior to the plant visit and the experience of the architectural engineer from previous similar activities.

The next step was a plant visit to verify the design and to collect further information to estimate costs relative to such factors as congestion, access, feasibility of equipment locations, and radiation levels. In practice the plant visit is essential and at ANO-1 improvements were made to the proposed modifications during the visit. Details that would be effectively impossible to obtain from available written material were obtained by observation and discussions with knowledgeable plant personnel.

Using the information collected during the site visit to improve the initial design, a final design was accomplished for each modification. Costs and radiation exposures associated with implementing the designs were then estimated using standard industry practices to produce the impact analysis. Details of the design and costs are given in Appendix J. A brief summary statement of the content of each alternative is given in Table 6.1.

6.2 Results

The results for the ANO-1 Plant are based in a set of economic ground rules and assumptions shown in Table 6.2 for both generic and local factors such as labor rates, replacement power costs per day, remaining economic life of the plant, owner's costs, and contingencies. The details of these factors are described in Appendix J. However, one example is owner's costs which consider factors such as retraining operators, health physics, quality assurance, review of contractor design, interaction with the NRC, and preparation of procedure and manual changes. Much of the detail available in Appendix J is provided for completeness of the impact estimates and possible use in any follow-on analyses. The basic information needed for the value-impact analysis in Section 9.0 is summarized in Table 6.3 using the local basis costs which is consistent with the use of local values in the value impact analysis.

The value-impact analysis variables shown in Table 6.3 (i.e., I_1 , I_3 , and V_3) reflect the fact that occupational dose is treated as a negative value rather than as a positive impact. Furthermore, the yearly occupational radiation dose (V_4) is also treated as a negative value but is considered to be negligible in the estimates for both the ANO-1 Alternatives. Similarly, the yearly replacement power costs (I_4) are considered to be negligible for ANO-1. Thus, Table 6.3 contains the necessary information to perform the value-impact analysis.

Although the local cost basis was used in the value-impact analysis, it is appropriate to note the ratio of the generic (in eastern Pennsylvania) to local engineering and installation costs which are:

Alternative	Generic/Local Cost Ratio
1	1.13
2	1.08

These ratios show that the generic costs are 13% greater for the alternative including only modifications to the existing plant and about 8% greater for the alternative involving the add-on dedicated shutdown decay heat removal system. (NOTE: If the values were computed for any representatiave site there would also be variations, but it cannot be stated a priori whether they would be greater or less.)

Table 6.1 Content of the Proposed Alternatives

Alternative 1 combines the following modifications:

a) Auxiliary Feedwater Pump Modification - This modification involves rewiring the existing auxiliary feedwater pump to both Class 1E trains and modifying the piping configuration to allow suction from the CST for use as an additional emergency feedwater pump. The purpose of this modification is to reduce the vulnerability of the emergency feedwater system to pump, turbine and battery failures.

The auxiliary feedwater pump is normally aligned to the Main Feedwater pump header and discharge to the normal feedwater supply header. This alignment allows use of the pump as a start-up feedwater pump during normal plant operation.

When required, the pump can now be used as an additional emergency feedwater pump by aligning the pump suction to the condensate storage tank. This realignment is accomplished by repositioning a motor-operated valve from a local control panel. The existing main feedwater control valves are used to control flow to the steam generators. Power to the pump is normally provided by the station A2 bus. If offsite power is not available, the pump can be powered by cross tying the A2 bus to the diesel generators. This will require stripping the other loads on the A2 bus.

b.) Addition of a Turbine Driven Generator for Emergency Loads - The turbine generator assembly for the proposed modification will be located outside of the auxiliary feedwater pump room north of existing pump P7B. Steam to drive the turbine will be supplied via a 4" pipeline connected to the existing main steam line that feeds pump P7A. A 4" globe valve in the new line provides manual isolation of the turbine. Turbine exhaust will be through a 12" pipe which discharges to the atmosphere.

The 500 kW generator will power a 480 volt substation (via the existing 4160V AC Bus A2) located near the turbine generator assembly. This substation then feeds the 480 volt MCC and the power and control transfer switches which are located in the control room. The battery chargers are provided with a separate manually operated transfer switch. Local and remote switch position indicators are provided for each transfer switch.

During normal plant operation, the turbine driven generator will be isolated from the main steam supply by means of a globe valve, and all equipment and circuits will be fed from the existing 4160 volt buses. If a station blackout occurs (loss of offsite power and DG failure), an operator will open the new manual globe valve allowing main steam to enter the turbine

Table 6.1 Content of the Proposed Alternatives (cont.)

steam chest which contains the turbine governor and trip valves. After the turbine generator has achieved operating speed, the operator will transfer the power feed for the battery chargers from the normal 480 volt bus to the new source to allow continued operation of the vital instrumentation.

The turbine generator is a constant speed device and is controlled by a skid-mounted governor and control valve. Operation of this unit simply requires opening a manual valve located near the turbine skid. Power transfer switches are also controlled locally with position indication (normal feed/emergency feed) provided locally and in the control room.

c) Addition of a Third DHR (low pressure) Pump - This modification consists of adding a third 100% DHR pump in parallel with the existing pumps. The pump will be located outside the existing ESF B Train pump room on elevation 317'0" and will be capable of replacing either existing DHR pump.

Alignment of the new pump is a manual operation. Local pump controls are provided with status lights in the control room. The pump shall be capable of being powered from either Class 1E distribution system. During normal plant operation, the new pump will be isolated by locked closed valves on the suction and discharge headers to maintain separation of the existing DHR trains. The new pump will be utilized only in the event of a loss of one of the existing pumps.

d) Redundant BWST Supply Valves - This modification provides a redundant flow path for safety injection by installing a parallel valve station in the main discharge line from the BWST. A manual valve is installed in a parallel line to the existing BWST discharge, upstream of two parallel air operated valves which feed one safety injection train in parallel with existing valves CV 1407A and CV 1408B. The new valves are installed in the pipe chase on the east end of the Auxiliary Building on elev. 335'.

The new manual valve is used only for maintenance and testing. It is locked in the open position during plant operation. Each new control valve will be capable of passing full safety injection design flow. The new air operated valves will receive a safety injection signal from the channels which feed the opposite train existing valve. That is the new CVA and existing CV 1408B will be powered from the same train and CVB and CV 1707A will be powered from the other train.

e) Improved Seismic Anchorage of Electrical Equipment - To add anchorage or strengthen existing anchors, steel angles will be welded to the base of electrical equipment. Holes will be drilled in the concrete floors and expansion anchors added. In cases where some anchorage is already provided, the existing

Table 6.1 Content of the Proposed Alternatives (cont.)

anchorage will be left in place because original, embedded anchorage is inherently stronger than expansion anchors.

4160 V Switchgear Cabinets

There are two switchgear cabinets in the switchgear Room which contain the Safeguard Buses. Each cabinet will be anchored along the twin sides by drilling and anchoring the existing channels to the concrete floor with expansion bolts.

480 V Transformer Buses

There are two 480 V transformer cabinets and two Motor Control Centers in the Switchgear Room. Each unit will be anchored by welding angles in the ends of the cabinet. The angles will be anchored to the concrete floor by expansion bolts.

Control Room panels

There are thirteen console and panel cabinets in the control room. Each cabinet will be anchored to the floor by either drilling through the existing angle braces or by adding channels to the frame to use with existing embedded anchors.

f) Condensate Storage Tank and Borated Water Storage Tank - This modification reduces plant vulnerability to a seismic event by adding braces to the existing condensate storage tank and borated water storage tank. This modification will increase the strength of the tanks sufficiently to ensure their availability following a seismic event.

g) Cable Spreading Room Fire Protection System - This modification will increase the reliability of the existing fire protection system actuation by adding a redundant deluge valve in the main header. The new valve will be activated by a new, separate sensing system and also will be controlled from a new local panel. The new deluge valve and isolation valves will be installed in parallel with the existing valves. There will be a total of 1000 ft of Detectowire for alarm and automatic actuation of the new valve. These will operate in conjunction with the existing detection systems.

As the presence of products of combustion in the vicinity of the trays, indicative of an incipient fire, is detected by any one of the existing smoke detectors, an alarm is sounded both locally and in the control room. The operator has the option of opening the appropriate deluge valve from the local panel or from the control room. Under normal circumstances, the operator will investigate the situation before taking action.

Table 6.1 Content of the Proposed Alternatives (cont.)

If a fire develops, the temperature sensitive Detectowire will automatically alarm and open the appropriate deluge valve. If, for any reason, the fire were to continue to develop, the area temperature detectors will alarm and open the appropriate new deluge valve. A periodic electrical continuity test of the deluge valve solenoid/fire detector circuits will be performed from the local control panel by pushing a button and observing a light.

There will be one three-zone fire protection control panel Pyrotronic "System 3" or a similar control panel, serving the new deluge valve. The control panel will include a 24 V battery which will supply the required power to the control circuits independently of the external AC power supply. The three-zone fire protection control panel will be located in a waterproof cabinet located outside the protected area. Electric power supply will be provided to the battery chargers in the local three-zone fire protection control panel. Alarm functions and remote control functions for this fire protection system will be added to the existing fire control cabinet in the main control room.

h) Diesel Generator Exhaust Stacks - There are two diesel generators which are located on the west side of the Auxiliary Building. For both diesel generators, the exhaust piping penetrates the west wall of the Diesel Generator Room and runs vertically upwards, parallel to the exterior face of that wall. There is currently one support for the vertical portion of this piping. The modification will consist of adding two structural steel strap supports per exhaust line on the unsupported vertical runs outside the building. The support will be attached to the adjacent, reinforced concrete wall with expansion bolts.

Alternative 2 - Add-on Decay Heat Removal System:

This modification consists of a single train of feedwater and primary makeup housed within a seismic Category I structure. Both the feedwater (two 6" supply lines and the primary makeup) and one 3" supply line are connected to the existing plant systems outside the containment.

The Add-on Decay Heat Removal System is completely self-contained within the ADHR tank building and ADHR structure. The ADHR tank building houses one 170,000 gal. borated water storage tank for primary makeup and one 200,000 gal. condensate storage for auxiliary feedwater. Both structures are seismic Category I, and in addition to housing the components identified herein, they provide missile protection (turbine and tornado). The tank building is connected to the ADHR structure by a 10-foot

Table 6.1 Content of the Proposed Alternatives (continued)

seismic Category I tunnel. This ADHR structure houses a single train motor driven auxiliary feedwater pump, a redundant train motor driven primary makeup pump, a diesel generator complete with all auxiliaries, and the power, control and service systems necessary for operation.

The normal power supply is offsite power via the switchyard and a 34.5/4.16 kV step-down transformer located outside the ADHR structure. Power and control distribution are provided by way of the 4160V and 480V switchgear located within the ADHR structure. The diesel generator is aligned for automatic start and load transfer on the loss of offsite power. Diesel cooling is provided through a closed loop service water system arranged to transfer jacket water and lube oil heat to the atmosphere via a service water to air heat exchanger (radiator) located within the ADHR. Diesel air intake and exhaust systems are protected from missiles as are the diesel room air cooling system inlet louvers and exhausts.

The ADHR tank building and structure are located southwest of Unit No. 1 inside the security fence. The ADHR structure is connected to the existing auxiliary building by a seismic Category I tunnel. This tunnel contains the 3" makeup and two 6" auxiliary feedwater lines in addition to power and control conduit and cable for the motor operated isolation valves, steam dump valves, and instrumentation.

The tunnels (ADHR tank building to ADHR structure and ADHR structure to auxiliary building Unit No. 1) are designed for man access for maintenance and inspection. These tunnels provide missile protection for the piping and conduits.

All piping and conduit within the add-on structures, tunnels, auxiliary building and containment are seismically supported with supports located on 10'0" centers.

The tunnel from the ADHR structure to the auxiliary building connects to the new tunnel between the new CST and the auxiliary building. Pipe and conduit penetrations are cut into the existing auxiliary building foundation wall and tunnel access provided by way of a locked manhole and ladder from the ADHR building grade floor.

Table 6.2 Economic Ground Rules and Assumptions
From Appendix J

<u>PARAMETER</u>	<u>GENERIC</u>	<u>LOCAL</u>
1. Plant Location	Eastern Pennsylvania	Northeast Arkansas
2. Remaining Economic Life (yrs)	19 yrs	22 yrs
3. Inflation Rate (%)	6	6
4. Discount Rate (%)	10.5	10.0
5. Fixed Charge Rate (%)	17.0	16.0
6. O&M Levelization Factor	1.47	1.69
7. Replacement Power Costs (\$/MW(e) - Day)	550(1)	423
8. Average Refueling Shutdown Length (Days)		
PWR	60	60
BWR	70	--
9. O&M Labor Rate (\$/Hr)(2)	22.80	16.02
10. O&M Labor/Materials Ratio	40/60	40/60
11. Owner's Costs (%)	10	10
12. Contingency (%)	25	25

Note: (1) MAAC Power Pool Average

(2) Includes employer's social security and
workmen's compensation contributions

Table 6.3 ANO-1 Impact Analysis Results

Alternative	Engineering and Installation Costs in 1985 Dollars \$ x 10 ⁻³	Operations and Maintenance Cost in 1985 Dollars \$ x 10 ⁻³	Occupational Dose During Installation in person-Rem*
1	14,298	80	0
2	54,697	353	20
Value-Impact Analysis Variable	I ₁	I ₂	V ₃

* The occupational dose for alternative 1 is substantially less than that for alternative 2 because there is no need to enter containment to make the modifications.

7.0 ALTERNATIVE VALUE ANALYSES

This section will describe the core melt probabilities and public risk estimates for the base case before any suggested modifications are applied and the improvements resulting from each alternative, i.e., combination of modifications.

7.1 Core Melt Probabilities

7.1.1 Base Case Probabilities

The core melt probability is the sum of the probabilities from the internal analysis and each of the special emergency analyses except for sabotage which is as yet not clearly quantifiable. The following list gives each probability and the section of this report where it was derived.

<u>Core Melt Probability</u>	<u>Reference Section</u>
Internal	8.8E-5
Seismic	7.3E-5
Fire	5.8E-6
Wind & Missiles	5.3E-6
External Flood	7.2E-6
Lightning	1.8E-7

1.79E-4 base core melt probability

7.1.2 Alternative Probabilities

The alternative probabilities are determined from the combination of internal and special emergency modifications included in that alternative. The value analysis for each modification is found in the applicable appendix. The results are summarized below:

<u>Modification of Vulnerability</u>	<u>Base Core Melt Probability</u>	<u>Core Melt Probability After Modification</u>
Internal 1	6.8E-5	1.2E-6 *
4	5.2E-6	3.9E-9 *
5	1.6E-5	6.6E-7 *
6	1.1E-5	2.8E-7 *
7	5.1E-6	1.7E-6 *
8	5.9E-6	1.5E-8 *
Seismic	7.3E-5	8.8E-6
Fire	5.8E-6	8.1E-7
Wind	5.3E-6	< 1.0E-7

* Estimated

The special emergency and internal analyses are essentially independent in the ANO-1 value analysis (there are possible

impact dependencies). This is not always true in PRA, but it is the case for ANO-1. The two alternatives selected will now be discussed and evaluated.

Alternative 1 Core Melt Probability - Alternative 1 is the combination of the internal modifications 1, 4, 5, 6, 7, and 8, seismic, fire, and extreme wind. The core melt probability of alternative 1 is the sum of the internal (with recovery) and external core melt probabilities with the above modifications. The relative importance of each vulnerability and the corresponding modification can be seen from the table above and from Table 5.1.

To compute each modification as an alternative all the way through to the risk measures would be unnecessarily expensive. Thus, the modifications are grouped as alternatives. However, for ANO-1 there didn't seem to be any good argument for more than one alternative to cover all the proposed modifications. The alternative 1 core melt probability is:

Internal (modifications 1 and 4-8)	3.44E-5
Seismic (modified)	8.8E-6
Fire (modified)	8.1E-7
Wind (modified)	1.0E-7
Lightning (no modifications)	1.8E-7
External Flood (no modifications)	<u>7.2E-6</u>

p(CM,alt 1) = 5.14E-5

Alternative 2 Core Melt Probability - Alternative 2 is the addition of a dedicated shutdown decay heat removal system as described in Section 4.0. As noted in Section 4, the reduction in core melt probability from the add-on system changes depending upon whether or not offsite power is available. Also, because the add-on is an independent system, the improvement in core melt probability is simply the product of the add-on system unreliability and the base case probability. The add-on system unreliability is 5.2E-2 if offsite power (OSP) is available and 1.0E-1 if offsite power is not available. Therefore, to properly apply the add-on system unreliability the core melt probability for internal events must be split into that when OSP is available and that when OSP is not available. In special emergencies from spray, fire, wind, and lightning it is assumed offsite power is available. For seismic it is assumed offsite power is not available. In the internal analysis situation the core melt probability is partitioned by the accident sequence applicability to offsite power availability. Alternative 2 with the add-on applied to the base case is as follows:

	<u>Base Case Probability</u>	<u>Add-on Unreliability</u>	<u>Alt 2 Core Melt Probability</u>
Internal - OSP available	6.48E-5	5.2E-2	3.37E-6
Internal - OSP not available	2.27E-5	1.0E-1	2.27E-6
Seismic - OSP not available	7.3E-5	1.0E-1	7.3E-6
Fire - OSP available	5.80E-6	5.2E-2	3.02E-7
Wind - OSP not available	5.31E-6	1.0E-1	5.31E-7
External Flood - OSP not available	7.20E-6	1.0E-1	7.20E-7
Lightning - OSP available 1/2 the time	1.80E-7	$\frac{5.2E-2 + 1.0E-1}{2}$	<u>1.37E-8</u>

Thus $p(\text{CM, alt 2}) = 1.45E-5$

7.1.3 Summary of Probabilistic Core Melt Estimates

The results of the two alternatives can be summarized by the core melt probabilities and change in core melt probability from the base value.

<u>Alternative</u>	<u>Probability of Core Melt</u>	<u>Change in Probability of Core Melt from Base Case</u>	<u>Factor of Improvement from Base Case Probability</u>
Base case	1.79E-4		
1	5.14E-5	1.28E-4	3.5
2	1.45E-5	1.65E-4	12.3

The core melt probabilities calculated compare well with the estimates used in Section 5.0 to select the alternatives.

The effect of an alternative can be viewed in several ways. One way is to compare the alternative core melt probability to some desired probability. A second approach is simply to examine the relative improvement over some base situation. For example, alternative 1 is approximately a factor of 3 improvement and alternative 2 is approximately a factor of 12 improvement from the base value. Further conclusions will be discussed in Sections 9.0 and 11.0.

7.2 Public Risk Estimates

In past PRAs several public risk measures were used (e.g., early deaths, early injuries, latent deaths, property damage). The measure used in this analysis is the averted offsite dose

resulting from the release of radioactive material to the atmosphere and subsequent dispersion out to a radius of 50 miles from the plant site. All other values (positive or negative) are considered in the impact analysis (Section 6.0).

The averted offsite dose in dollars is calculated for the internal analysis and each special emergency (for ANO-1 these are seismic, external flood, fire, and wind and missiles).

7.2.1 Base Case Estimates

The base case estimate of the public risk measure (population dose) is computed from the base case probabilities for each accident sequence including containment systems, the containment failure mode probabilities and release category assignments, and the consequences for each release category. The base case internal analysis accident sequences with containment systems are found in Table B.27. The base case special emergency sequence probabilities are summarized from the various appendices in Table 7.1. The containment failure mode (CFM) probabilities and release category assignments for each type containment systems sequence are summarized in Table 7.2.

The results are tabulated in Table 7.3 and shown by dominant accident sequence in Table 7.4 in a format similar to that used in past PRAs, e.g., WASH 1400. The release category (RC) probabilities are then combined with the consequences determined in Appendix K to obtain the expected population dose in person-rem per reactor-year and cost in dollars (at \$1000 per person-rem) in Table 7.5. Three values are presented in Table 7.5: an upperbound for purposes of this study which is based upon WASH 1400 type source terms, our central estimate which we have defined as 0.3 times the WASH 1400 source term, and a lower bound which is defined as 0.1 times the WASH 1400 source term. This selection of source terms should not be interpreted as an endorsement of any particular set. The "real" source term may be larger or smaller. These were selected in order to provide some indication of the sensitivity of these results to variations in the source term.

Two examples may be useful to show how the calculation was done. First, consider the internal accident sequence $S_3MH_1H_2'$. This is the first sequence found in Table B.27 where it is seen that the $\bar{Z}_L\bar{C}\bar{F}$ containment sequence is most significant. Its probability is $1.52E-5$, i.e.,

$$p(S_3MH_1H_2' - \bar{Z}_L\bar{C}\bar{F}) = 1.52E-5.$$

This is a late melt sequence resulting from the ECR failure (i.e., H_1H_2'). Going to Table 7.2 this falls in the first

(Text continued on page 7-10.)

Table 7.1 ANO-1 Special Emergency Accident
Sequence Probabilities Including
Containment Systems

<u>Special Emergency</u>	<u>Containment Systems Sequence</u>	<u>Base Case Probability</u>	<u>Probability After Modification</u>
Fire	$Z_T C$	5.80E-5	8.10E-7
Wind	$Z_T C$	5.31E-6	~ 1.00E-7
Seismic	$\bar{Z}_L \bar{C} \bar{F}$ or $\bar{Z}_T \bar{C}$	1.42E-5	4.63E-7
Seismic	$\bar{Z}_L \bar{C} F (LM)$	NA	2.05E-7
Seismic	$\bar{Z}_L C$ or $\bar{Z}_T C$	2.17E-5	4.33E-7
Seismic	$Z_L \bar{C} \bar{F}$	2.13E-6	1.40E-9
Seismic	$Z_L C$ or $Z_T C$	3.50E-5	7.73E-6
Lightning	$Z_T C$	1.80E-7	1.80E-7
Ext. Flood	$Z_T C$	7.20E-6	7.20E-6

Table 7.2 Containment Sequences to
Release Category Mapping

Containment Systems Sequence	Special Conditions Core Melt	Containment Failure Mode with Probability and Release Category				
		α	β	γ, δ_e	δ_l	c
$\bar{Z}_L \bar{C}\bar{F}$ or $\bar{Z}_T \bar{C}$		1E-4 1	2E-3 5	1.4E-2 3	1.8E-1 5	2.5E-1 7
$\bar{Z}_L \bar{C}\bar{F}$	EM	1E-4 1	2E-3 5	1.4E-2 3	1.8E-1 4	2.5E-1 6
	LM	1E-4 1	2E-3 4	1.4E-2 2	1.8E-1 3	2.5E-1 6
$\bar{Z}_L \bar{C}$ or $\bar{Z}_T \bar{C}$		1E-4 1	2E-3 4	1.4E-2 2	1.8E-1 3	2.5E-1 6
$Z_L \bar{C}\bar{F}$		1E-4 1	2E-3 5	1.4E-2 3	1.8E-1 4	2.5E-1 7
$Z_L \bar{C}\bar{F}$	EM	1E-4 1	2E-3 5	1.4E-2 3	1.8E-1 4	2.5E-1 7
	LM	1E-4 1	2E-3 4	1.4E-2 2	1.8E-1 3	2.5E-1 7
$Z_L \bar{C}$ or $Z_T \bar{C}$		1E-4 1	2E-3 4	1.4E-2 2	1.8E-1 3	2.5E-1 6

TABLE 7.3 ANO-1 - RELEASE CATEGORY SUMMARY - BASECASE (All Probabilities are per Reactor-Year)

Accident Sequence	Prob.	RC 1		RC 2		RC 3		RC 4		RC 5		RC 6		RC 7	
		Mode	Prob.	Mode	Prob.	Mode	Prob.	Mode	Prob.	Mode	Prob.	Mode	Prob.	Mode	Prob.
INTERNAL- $\bar{Z}_L \bar{C}F$ or $\bar{Z}_T \bar{C}$	3.57E-5	α	3.57E-9			γ, δ_e	5.00E-7			β, δ_k	6.43E-6			ϵ	8.93E-6
INTERNAL- $\bar{Z}_L \bar{C}F$ (LM)	1.30E-5	α	1.30E-9	γ, δ_e	1.82E-7	δ_k	2.34E-6	β	2.60E-8			ϵ	3.25E-6		
INTERNAL- $\bar{Z}_L C$ or $\bar{Z}_T C$	1.25E-5	α	1.25E-9	γ, δ_e	1.75E-7	δ_k	2.25E-6	β	2.50E-8			ϵ	3.13E-6		
INTERNAL- $Z_L \bar{C}F$	3.76E-8	α	3.76E-12			γ, δ_e	5.26E-10	δ_k	6.77E-9	β	7.52E-11			ϵ	9.40E-9
INTERNAL- $Z_L \bar{C}F$ (LM)	2.67E-8	α	2.67E-12	γ, δ_e	3.74E-10	δ_k	4.81E-9	β	5.34E-11					ϵ	6.68E-9
INTERNAL- $Z_L C$ or Z_T	2.47E-5	α	2.47E-9	γ, δ_e	3.46E-7	δ_k	4.45E-6	β	4.94E-8			ϵ	6.18E-6		
FIRE- Z_T	5.80E-6	α	5.80E-10	γ, δ_e	8.12E-8	δ_k	1.04E-6	β	1.16E-8			ϵ	1.45E-6		
WIND- Z_T	5.31E-6	α	5.31E-10	γ, δ_e	7.43E-8	δ_k	9.56E-7	β	1.06E-8			ϵ	1.33E-6		
LIGHTNING- Z_T	1.80E-7	α	1.80E-11	γ, δ_e	2.52E-9	δ_k	3.24E-8	β	3.60E-10			ϵ	4.50E-8		
EXT FLOOD- Z_T	7.20E-6	α	7.20E-10	γ, δ_e	1.01E-7	δ_k	1.30E-6	β	1.44E-8			ϵ	1.80E-6		
SEISMIC- $\bar{Z}_L \bar{C}F$ or $\bar{Z}_T \bar{C}$	1.42E-5	α	1.42E-9			γ, δ_e	1.99E-7			β, δ_k	2.58E-6			ϵ	3.55E-6
SEISMIC- $\bar{Z}_L C$ or $\bar{Z}_T C$	2.17E-5	α	2.17E-9	γ, δ_e	3.04E-7	δ_k	3.91E-6	β	4.34E-8			ϵ	5.43E-6		
SEISMIC- $Z_L \bar{C}F$	2.13E-6	α	2.13E-10			γ, δ_e	2.98E-8	δ_k	3.83E-7	β	4.26E-9			ϵ	5.33E-7
SEISMIC- $Z_L C$ or Z_T	3.50E-5	α	3.50E-9	γ, δ_e	4.90E-7	δ_k	6.30E-6	β	7.00E-8			ϵ	8.75E-6		
TOTAL			1.78E-8		1.76E-6		2.33E-5		5.78E-7		9.01E-6		3.14E-5		1.30E-5

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TABLE 7.4 ANO-1 DOMINANT ACCIDENT SEQUENCES AND SPECIAL EMERGENCIES CONTRIBUTING TO RELEASE CATEGORIES

		RELEASE CATEGORIES						
		1	2	3	4	5	6	7
S T E M M A V A N G I T Y I T I M I	S _{2,3} LOCA	S ₃ MH ₁ H ₂ ¹ -Z _L C ⁻ α 1.52E-9 S ₃ MXE-Z _L C ⁻ α 2.12E-10 S ₃ MED ₂ -Z _L C-α 8.05E-11	S ₃ MED ₂ -Z _L C-γ, δ _e 1.13E-8	S ₃ MH ₁ H ₂ ¹ -Z _L C ⁻ δ _k 1.46E-6 S ₃ MED ₂ -Z _L C-δ _k 1.04E-6 S ₃ MED ₂ -Z _L C-δ _k 1.45E-7	S ₃ MED ₂ -Z _L C-B 1.61E-9 S ₃ MH ₁ H ₂ ¹ -Z _L C ⁻ δ _k 1.62E-8 S ₃ MED ₂ -Z _L C-B 1.16E-8	S ₃ MH ₁ H ₂ ¹ -Z _L C ⁻ δ _k 2.77E-6 S ₃ MXE-Z _L C ⁻ δ _k 3.86E-7	S ₃ MED ₂ -Z _L C-ε 2.01E-7 S ₃ MH ₁ H ₂ ¹ -Z _L C ⁻ δ _k 2.02E-6 S ₃ MXE-Z _L C ⁻ ε 5.30E-7	S ₃ MH ₁ H ₂ ¹ -Z _L C ⁻ δ _k 3.80E-6 S ₃ MXE-Z _L C ⁻ ε 5.30E-7
	T ₁ TRANSIENT	T ₁ MLE-Z _T -α 2.02E-9 T ₁ MQLD ₁ -Z _L C-α 7.48E-11	T ₁ MLE-Z _T -γ, δ _e 2.83E-7 T ₁ MQLD ₁ -Z _L C-γ, δ _e 1.05E-8	T ₁ MLE-Z _T -δ _k 3.64E-6 T ₁ MQLD ₁ -Z _L C-δ _k 1.35E-7	T ₁ MLE-Z _T -B 4.04E-8 T ₁ MQLD ₁ -Z _L C-B 1.50E-9		T ₁ MLE-Z _T -ε 5.05E-6 T ₁ MQLD ₁ -Z _L C-ε 1.87E-7	
	T ₂ TRANSIENT	T ₂ MLE-Z _T -α 1.51E-10 T ₂ MLE-Z _T -α 7.81E-11	T ₂ MLE-Z _T -γ, δ _e 1.09E-8	T ₂ MOD ₁ D ₂ -Z _L C-δ _k 1.02E-7 T ₂ MLE-Z _T -δ _k 1.41E-7	T ₂ MLE-Z _T -B 1.55E-9 T ₂ MOD ₁ D ₂ -Z _L C-B 1.13E-9	T ₂ MLE-Z _T -δ _k 2.75E-7	T ₂ MOD ₁ D ₂ -Z _L C-ε 2.43E-7	T ₂ MLE-Z _T -ε 3.88E-7
	T _{3,4,5} TRANSIENTS	T ₃ QD ₁ D ₂ -Z _L C-α 9.70E-11 T ₃ QH ₁ H ₂ ¹ -Z _L C ⁻ α 8.41E-10 T ₄ MLE-Z _T -α 3.17E-10 T ₅ MLE-Z _T -α 1.75E-10	T ₃ QD ₁ D ₂ -Z _L C-γ, δ _e 1.36E-8	T ₃ QD ₁ D ₂ -Z _L C-δ _k 1.75E-7 T ₃ QH ₁ H ₂ ¹ -Z _L C ⁻ δ _k 7.27E-7 T ₃ QD ₁ D ₂ -Z _L C-δ _k 6.97E-7 T ₄ MLE-Z _T -δ _k 1.34E-7	T ₃ QD ₁ D ₂ -Z _L C-B 7.74E-9 T ₃ QD ₁ D ₂ -Z _L C-B 1.94E-9 T ₃ QH ₁ H ₂ ¹ -Z _L C ⁻ δ _k 8.08E-9 T ₄ MLE-Z _T -B 1.48E-9	T ₃ QH ₁ H ₂ ¹ -Z _L C ⁻ δ _k 1.53E-6 T ₄ MLE-Z _T -δ _k 5.77E-7 T ₅ MLE-Z _T -δ _k 3.19E-7	T ₃ MOD ₁ D ₂ -Z _L C-ε 2.43E-7 T ₃ QH ₁ H ₂ ¹ -Z _L C ⁻ ε 1.01E-6	T ₃ QH ₁ H ₂ ¹ -Z _L C ⁻ ε 2.10E-6 T ₄ MLE-Z _T -ε 7.93E-7 T ₅ MLE-Z _T -ε 4.38E-7
	SPECIAL EMERGENCIES	SEISMIC-Z _T -α 1.42E-9 SEISMIC-Z _T /Z _L C-α 2.17E-9 SEISMIC-Z _T /Z _L C-α 3.50E-9	SEISMIC-Z _T /Z _L C-γ, δ _e 3.04E-7 SEISMIC-Z _T /Z _L C-γ, δ _e 4.90E-7	SEISMIC-Z _T /Z _L C-δ _k 3.91E-6 SEISMIC-Z _T /Z _L C-δ _k 6.30E-6 FIRE-Z _T -δ _k 1.04E-6 EXT FLOOD-Z _T -δ _k 1.30E-6	SEISMIC-Z _T /Z _L C-B 4.34E-8 SEISMIC-Z _T /Z _L C-B 7.00E-8 EXT FLOOD-Z _T -B 1.44E-8	SEISMIC-Z _T -δ _k 2.58E-6 SEISMIC-Z _T -ε 5.43E-6 FIRE-Z _T -ε 1.45E-6 WIND-Z _T -ε 1.33E-6 EXT FLOOD-Z _T -ε 1.80E-6	SEISMIC-Z _T -ε 5.43E-6 FIRE-Z _T -ε 1.45E-6 WIND-Z _T -ε 1.33E-6 EXT FLOOD-Z _T -ε 1.80E-6	SEISMIC-Z _T -ε 5.43E-6 FIRE-Z _T -ε 1.45E-6 WIND-Z _T -ε 1.33E-6 EXT FLOOD-Z _T -ε 1.80E-6
SUM		1.78E-8	1.76E-6	2.33E-5	5.78E-7	9.01E-6	3.14E-5	1.30E-5

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Table 7.5 ANO-1 Value of Offsite Population Dose
in Person-Rem for the Base Case
(person-rem/reactor-year)

<u>Release Category</u>	<u>Population Dose (PR) Within 50 Miles, D</u>	<u>Probability of Release Category, P</u>	<u>DP</u>
Upper Bound Source Term (WASH 1400)			
1	7.9E+5	1.8E-8	0.0
2	7.5E+5	1.8E-6	1.4
3	5.9E+5	2.3E-5	13.6
4	3.2E+5	5.8E-7	0.2
5	2.0E+5	9.0E-6	1.8
6	5.4E+4	3.1E-5	1.7
7	6.0E+3	1.3E-5	0.1
			18.8 PR
Baseline Source Term (0.3 times WASH 1400)			
1	4.7E+5	1.8E-8	0.0
2	4.9E+5	1.8E-6	0.9
3	3.5E+5	2.3E-5	8.1
4	2.1E+5	5.8E-7	0.1
5	1.2E+5	9.0E-6	0.1
6	2.3E+4	3.1E-5	0.7
7	2.4E+3	1.3E-5	0.0
			10.9 PR
Lower Bound Source Term (0.1 times WASH 1400)			
1	2.9E+5	1.8E-8	0.0
2	3.5E+5	1.8E-6	0.6
3	2.3E+5	2.3E-5	5.3
4	1.4E+5	5.8E-7	0.1
5	6.5E+4	9.0E-6	0.6
6	1.0E+4	3.1E-5	0.3
7	2.1E+3	1.3E-5	0.0
			6.9 PR

row of the table where 5 containment failure modes (CFM) are possible. As an illustration consider the most likely CFM which is ϵ with a probability of $2.5E-1$ and a release category 7 (RC7) assignment. Thus,

$$p(S_3 M H_1 H_2' - \bar{Z}_L \bar{C} F - \epsilon) = (1.52E-5) * (2.5E-1) = 3.80E-6.$$

This sequence is placed in row 1 (S_3 initiating event) column 7 (RC7) in Table 7.4. The initiating event segregation in Table 7.4 is not important for any reason except perhaps to make it easy to locate a sequence. This same procedure is repeated for every significant internal accident sequence.

Next consider a seismic special emergency sequence with the associated containment systems sequence $Z_L C / Z_T$. There are two seismic possibilities $Z_L C$, and Z_T . Its probability from Table 7.1 is $3.50E-5$, i.e.,

$$p(\text{SEISMIC} - Z_L C / Z_T) = 3.50E-5,$$

In this case let us consider all the CFMs. These are:

$$p(\text{SEISMIC} - Z_L C / Z_T - \alpha) = (3.50E-5) * (1E-4) = 3.50E-9 \text{ RC1},$$

$$p(\text{SEISMIC} - Z_L C / Z_T - \beta) = (3.50E-5) * (2E-3) = 7.00E-8 \text{ RC4},$$

$$p(\text{SEISMIC} - Z_L C / Z_T - \gamma, \delta_e) = (3.50E-5) * (1.4E-2) = 4.90E-7 \text{ RC2},$$

$$p(\text{SEISMIC} - Z_L C / Z_T - \delta_g) = (3.50E-5) * (1.8E-1) = 6.30E-6 \text{ RC3}.$$

$$\text{and } p(\text{SEISMIC} - Z_L C / Z_T - \epsilon) = (3.50E-5) * (2.5E-1) = 8.75E-6 \text{ RC6}.$$

When these are entered into Table 7.4, it is seen that this seismic sequence contributes significantly to all five release categories. Again, the example serves to demonstrate the mapping of sequences into release categories. Actually, each sequence could also be carried on to the population dose but it is easier to combine all the sequences in the release categories.

7.2.2 Alternative Estimates

The population dose estimates for the alternatives are determined by the same procedure used for the base case. The tables containing the information needed are:

Table

Alternative 1	Internal B.30
Alternative 2	Internal B.31

Table

Special Emergencies 7.1
Special Emergencies 7.1

In each alternative the applicable special emergencies must be considered:

Alternative 1	Seismic, Fire, Wind, Lightning, External Flood
Alternative 2	All Special Emergencies are effectively covered

The results comparable to those obtained for the base case are found in Tables 7.6 and 7.7.

7.2.3 Summary of Probabilistic Risk Estimates

The results of averted offsite population dose per reactor year are summarized in Table 7.8 for the baseline source term and the upper and lower bounds.

7.3 Non-Quantifiable Values

The previous sections have outlined the value of the potential alternatives based upon probabilistic considerations. As was noted earlier, these alternatives were selected using the systematic PRA approach, however the scope of the modifications is such that they also provide value which is in many respects non-quantifiable. These potential values include such diverse areas as: the overall effect such alternatives may have on residual risk, the potential effect of at least some of them on equipment qualification issues and fire (Appendix R) concerns, and the possible effects on plant availability. Other issues which may well be affected are regulatory stability and decisions as to the need for changes to decay heat removal requirements. Of course, the primary objective of TAP A-45 is, in fact, to evaluate the adequacy of existing requirements. These various areas are discussed further in the following subsections.

7.3.1 Reductions in Residual Risk

It was noted in Section 5.0 that the actual total core melt probability, which would include large LOCAs and ATWS, is greater than the base case numbers shown for transients and SBLOCAs. For those events which are quantified there is a threshold of probability that establishes a level below which events are "not counted". This procedure is dictated in part by the capabilities of existing analytical tools. However, many of these "uncounted" events may be affected by the support system vulnerabilities such as are identified here. Therefore, improvements in support systems (e.g., emergency electrical power) such as those provided in alternative 1 will (should) reduce core melt probability even more than is shown here.

**Table 7.6 ANO-1 Release Category Probabilities
(Probability per reactor-year)**

	RC1	RC2	RC3	RC4	PC5	RC6	RC7
Base Case	1.8E-8	1.8E-6	2.3E-5	5.8E-7	9.0E-6	3.1E-5	1.3E-5
Alternative 1	5.2E-9	5.4E-7	7.2E-6	8.0E-8	2.5E-6	9.6E-6	3.4E-6
Alternative 2	1.4E-9	1.5E-7	2.0E-6	5.4E-8	6.0E-7	2.7E-6	8.8E-7

**Table 7.7 ANO-1 Expected Value of the Population Dose
Within 50 Miles (person-rem per reactor year)**

	Lower Bound Source Term (0.1 x WASH 1400 Source Term)	Baseline Source Term (0.3 x WASH 1400 Source Term)	Upper Bound Source Term (WASH 1400 Source Term)
Base Case	6.9	10.9	18.8
Alternative 1	2.2	3.3	5.7
Alternative 2	0.6	1.0	1.6

**Table 7.8 ANO-1 Summary of Values for Averted Offsite
Population Dose in Person-Rem per Reactor Year
Including Internal and Special Emergencies**

	<u>Lower Bound Source Term</u>	<u>Baseline (Central) Source Term</u>	<u>Upper Bound Source Term</u>
Alternative 1	4.7	7.6	13.1
Alternative 2	6.3	9.9	17.2

Furthermore, since some of the modifications address potential vulnerabilities in support systems, there also will be improvements in the response to other initiating events such as large LOCAs which have not been addressed here. Furthermore, any overall improvement in safety systems reliability will obviously reduce risk even though it may not be quantified.

It should be noted that the add-on system has a significant effect upon residual risk, whatever the source of that risk. This beneficial effect arises from the fact that the add-on is independent from other plant systems except for the required tie-ins inside containment. Therefore, the add-on is unaffected by any fires or floods which might occur elsewhere in the plant. As noted earlier, the unreliability of the add-on is a function of the availability of offsite power (OSP), and this unreliability becomes a multiplicative factor, i.e., it reduces core melt probability. The single train described here has unreliabilities of $5.2E-2$ with OSP available, and $1.0E-1$ without OSP. If a two train system were used in which complete independence between trains was maintained, then these unreliabilities become $2.7E-3$ and $1.0E-2$ respectively, providing an even greater reduction in residual risk. However, there would be an accompanying increase in costs.

7.3.2 Effects Upon Equipment Qualification

A subject of concern in many current discussions is equipment qualification. That is, is the equipment truly qualified to withstand the environments it may experience under accident conditions. The question arises, for example, if one considers primary system feed and bleed options for decay heat removal (see Section 10.0). It also arises if one considers piping failures that might fill a room or compartment with high temperature water or steam, or both. Similar concerns have been expressed by some with respect to inadvertent actuation of fire suppression systems that use water spray. Some of the modifications proposed in the above alternatives do deal with equipment qualification issues. For example, the seismic based modifications which provide additional tie down of electrical cabinets is intended to improve system survivability in an earthquake environment. The add-on system provides an alternative approach to dealing with equipment qualification concerns. As noted above, a completely independent train of equipment can be (and is, in this design) isolated from adverse environments that might occur in the various plant buildings. Also, the separation into a dedicated building allows more stringent controls to insure that adverse environments are not created within that building. The dedicated system does not solve environmental qualification problems, if they exist, but it does reduce their impact and influence.

7.3.3 Effects Upon Plant Availability

All the modifications proposed here have improvement of decay heat removal as the goal. It is clear that such modifications could affect in-service plant availability in either a positive (improved) and negative (decreased) manner. A modification which adds equipment to a system could possibly lead to an increase in the spurious scram rate, and thus decrease availability. Or in other situations, the addition of equipment could lead to a situation in which failure of existing components with a similar function are covered by the added redundancy or diversity. Thus, in some situations, plant operation could continue without interruption, that is, technical specifications would be met while equipment was repaired rather than having to shut down while the work was being done. It seems self-evident that if the additional equipment is in the form of a standby system the probability of spurious scrams should be small. Certainly the frequency of those situations for which additional equipment could obviate a plant shutdown would be dependent upon the design of the plant and the nature of the added equipment. This area could be explored further in subsequent analyses.

7.3.4 Regulatory Issues

It has been suggested that revisions to decay heat removal capabilities could result in relaxation of requirements and perhaps contribute to increased regulatory stability. These issues should be pursued as part of the overall TAP A-45 program. However, the proper forum or agency for those evaluations is not as obvious. Certainly the technical analyses bear upon these issues, but they also require policy and regulatory perspectives which are more properly the purview of the NRC. It is also apparent that these issues should not (and perhaps cannot) be resolved on the basis of one plant study. This question also will be pursued further in subsequent analyses.

7.3.5 Summary of Non-Quantifiable Values

Based upon the work to date, it is quite apparent that the modifications being proposed can reduce risks beyond that quantified here. It is equally obvious that some alternatives such as the independent add-on system provide another means to address equipment qualification issues. The effect of DHR modifications upon plant availability will be plant specific. Several regulatory issues can also be affected by DHR modifications but definitive discussion requires added input in terms of plant analyses and NRC regulatory policy.

8.0 UNCERTAINTY AND SENSITIVITY CONSIDERATIONS

In this chapter the sources of uncertainty in the estimates of the "Impact" and "Value" terms discussed in Chapters 6 and 7, respectively, are examined and, where possible, quantified so that the overall uncertainty in the integrated value-impact analysis described in Chapter 9 can be evaluated.

It should be noted that in this chapter the avertable onsite costs, which are treated as "negative impacts" in Chapters 6 and 9, are treated as "positive values" as this facilitates assessment of the overall uncertainty in the value-impact analysis.

It should also be noted that there are a number of parameters for which the numerical values depend on decisions by NRC; the manner in which the value-impact ratios (see Appendix L) would vary with these parameters is indicated (e.g., whether linear or otherwise) so that the decision maker can readily examine the effect of changes in the values currently preferred by NRC.

8.1 The Nature of the Problem

It has been shown in a generic study of the treatment of uncertainty in value impact analysis³⁰ that a useful approach is as follows:

- i) By treating avertable onsite costs as positive values instead of negative impacts, the uncertainty in the impact term can be ignored since compared to the uncertainty in the value term it is very small.
- ii) By analysis of the structure of the value term it can be seen which parameters can be regarded as subject to random variations, and thus can be treated statistically, and which should be regarded as empirical choices that would best be treated by means of a sensitivity analysis.
- iii) If the value term is limited to offsite costs, the uncertainty can be large (see below), but if it includes onsite costs as well, the uncertainty in the results of the analysis is reduced. This is because the uncertainty in the onsite costs is smaller than that of the offsite costs and, in general, the magnitude of the onsite costs is considerably greater; consequently the large uncertainty in the offsite cost becomes less important. The current views of the NRC Steering Committee of the Safety Goals Evaluation Program on the inclusion of onsite costs are summarized in Reference 29.

In the case of a nuclear power plant for which a PRA exists that includes an estimate of the consequences, the relative magnitudes of the offsite and onsite costs can be determined

directly. Consequently, if the value term includes both onsite and offsite costs, a better appreciation of the overall uncertainty in the results of the analysis can be obtained than is possible by means of a generic approach. The availability of a plant specific PRA may also make possible a better estimate of the uncertainty in some individual parameters of the value terms, such as the probability of core melt.

As discussed in the next section, in the case of the ANO-1 NPP, the number of sequences which make a substantial contribution to the overall risk is small. This facilitates examination of the amount of the overall uncertainty attributable to the uncertainty in the probability of core melt, $p(m)_j$, due to accident sequences j , and in q_{jk} , the conditional probability of a release in category k , given core melt due to accident sequence j .

8.2 The Uncertainty in the Offsite Costs

8.2.1 The structure of the offsite cost terms

It is shown in Reference 30 that if offsite costs are evaluated in terms of $\$C(D)$ per person rem of population dose, where $C(D)$ is in the range \$10 to \$1000 per person rem, the offsite cost term, $V(T)$, which is the present worth of the expectation of loss, can be represented by the following expression:

$$V(T) = n_{\text{eff}} C(D) \sum_{\text{all } k} F(D)_k \sum_{\text{all } j} p(r)_{jk} \quad (1)$$

where n_{eff} is the discounted, or effective, remaining life of the plant in years.

$F(D)_k$ is a function which converts a release in category k into population dose, in averaged weather conditions, and $p(r)_{jk}$ is the probability of a release of category k due to accident sequence j . Also, $p(r)_{jk} = p(m)_j \cdot q_{jk}$ where $p(m)_j$ is probability of core melt due to sequence j and, as noted above, q_{jk} is conditional probability of a category k release, given core melt due to sequence j ; it should be noted that the definition of q_{jk} is NOT simply the probability of containment failure, given core melt due to sequence j . However, it may have the same numerical value.

It is also shown in Reference 30 that the change in offsite costs $\Delta V(T)$, due to an improvement which reduces $p(m)_j$ by $\Delta p(m)_j$ and q_{jk} by Δq_{jk} can be written

$$\Delta V(T) = n_{\text{eff}} C(D) \sum_{\text{all } k} F(D)_k \sum_{\text{all } j} [(q_{jk} \Delta p(m)_j + p(m)_j \Delta q_{jk})] \quad (2)$$

The uncertainties in each of the terms on right hand side of Equation (2) are discussed in the following Sections. (Note:

A further problem, arising from the fact that both $p(m)$ and q may contain unquantifiable components is discussed in Reference 31.)

8.2.2 The uncertainty in n_{eff}

With the assumption that the remaining life of the plant, N , is its design life less the current number of years of commercial operation, the only uncertainty in n_{eff} is the choice of discount rate.

In the case of ANO-1 N is 22 years (for both units) and n_{eff} is

13.2 years for a discount rate of 5% and
8.8 years for a discount rate of 10%.

The choice of discount rate is at NRC's discretion, 5% appears to be the preferred value at the present time and this is the value used in this report, unless otherwise stated.

8.2.3 The uncertainty in $C(D)$

NRC's preferred value for $C(D)$ is \$1000 per person rem of population dose, based on the total dose out to a distance of 50 miles.³² This choice of $C(D)$, or the distance to which population dose is integrated, might conceivably change; several studies of alternative methods for deriving the monetary value of the consequences of severe accidents are described in the literature.^{33,34} The extent to which estimates based on the NRC's current guideline values for $C(D)$ and distance differ from those derived by more complex models is discussed in Reference 30. Conceivable changes in NRC's ground rules could change the estimated monetary values by a factor of about 3 upwards or downwards. Reasonable sets of assumptions used with the more complex models could increase the estimates derived in Reference 34 by a factor of about 5, as described in Reference 31.

Clearly, the choice of $C(D)$ and the distance to which population dose is integrated are at NRC's discretion, and consequently does not need to be considered as a source of uncertainty.

8.2.4 The uncertainty in $F(D)$

In addition to the uncertainty arising from the choice of $C(D)$ there is also some uncertainty in the conversion of a release of any given size and composition to population dose. This is discussed in Reference 30, where it is shown that application of the interdiction dose concept largely offsets most of the uncertainties arising from factors such as atmospheric dispersion, dry and wet deposition, buoyancy effects in the

plume (if the population distribution is reasonably uniform) and the cost and effectiveness of decontamination processes. Consequently, the uncertainty in population dose due to these factors is small, given a reasonably uniform population distribution, although the uncertainty in property damage, due to decontamination and interdiction costs may be large.

In addition to the uncertainty in the conversion of a given category of release into population dose, there is a further uncertainty as to the correct values of the release fraction for the various isotopes occurring under the conditions which characterize that particular category. This aspect is discussed in Reference 30, where it is concluded that the fractional releases shown in Table 2.3 of Appendix K to this report may be regarded as upper bounds; for the purposes of obtaining central values for $V(T)$ it is considered that release fractions which are one third of those shown in Table 2.3 would be a more appropriate choice and fractions which are one tenth of those in Table 2.3 could be regarded as lower bounds.

However, from the results obtained using the CRAC2 program³⁵ it appears that when population dose is integrated over a radius of only 50 miles, with an interdiction dose of 25 rem over 30 years, the total dose is remarkably insensitive to changes in the release fractions within the suggested upper and lower bounds. It can be seen from Table 2.4, Appendix K, of this report, that the variation about the central value for population dose is covered by a factor of 1.5 for Category 2, which is the dominant contributor to risk. Consequently, the effect of this variation on the overall uncertainty is negligible.

In the specific case of ANO-1, the population distribution is somewhat asymmetrical, due to the lakeside location of the plant, and on the landward side the population density, out to 30 miles, in the most densely populated sector is about 5 times the average for the other landward sectors. Thus, if a large release were to occur, the actual population dose, out to 50 miles could be substantially greater than that predicted for averaged weather conditions, depending on the wind direction at the time of the release.

Thus, if decision making can be based on averaged weather conditions, the uncertainty in $F(D)$ is small and can be ignored. However, if it were considered necessary to take account of day to day weather variations, $F(D)$ could vary by a factor of about 2 with respect to the central value. It is assumed in this analysis that NRC's decisions would be based primarily on averaged weather conditions.

The two ANO reactors (units) do not share safety systems, and therefore, there are few, if any, accident sequences for ANO which could lead to core melt in both units. If such sequences

did exist, the source terms for any given release category would be doubled. However, it can be seen from Tables 2.3 and 2.4 of Appendix K and the preceding discussion that this would have only a small effect, of order 10%, on population dose out to 50 miles. Thus the additional uncertainty in the offsite costs due to this possible increase in source terms is negligible.

8.2.5 The uncertainty in $P(m)$ and q

From Equation (2), above, it can be seen that if a modification changes $p(m)_j$ without affecting q_{jk} (i.e., $\Delta q_{jk} = 0$). The equation reduces to

$$\Delta V(T) = n_{\text{eff}} C(D) \sum_{\text{all } k} F(D)_k \sum_{\text{all } j} q_{jk} \Delta p(m)_j \quad (3)$$

It is shown in Reference 35 that if the fractional improvement in q_{jk} is the same as that in $p(m)_j$, Eqn (2) becomes

$$\Delta V(T) = n_{\text{eff}} C(D) \sum_{\text{all } k} F(D)_k 2 \sum_{\text{all } j} q_{jk} \Delta p(m)_j \quad (4)$$

The two cases, $\Delta q_{jk} = 0$ and $\Delta q_{jk}/q_{jk} = \Delta p(m)_j/p(m)_j$ should bound the situations arising in practice; thus Equations (3) and (4) can be used to examine the effect of uncertainties in $\Delta p(m)$ and q .

From Table 7.5 of Section 7 of this report, it can be seen that most of the population dose (83%) is attributable to events giving rise to release in Categories 2 and 3 and that, of this contribution, about 47% is attributable to "internal" initiating events. About 44% of the release in Categories 2 and 3 is due to seismic events; four percent is due to fire and five percent due to external flood.

From previous PRAs it appears likely that the uncertainty in the estimated values for the probability of core melt due to internal initiating events which arises from random variations can be represented by an error factor* of about 3; the estimated probabilities for releases due to external initiating events are larger, an error factor of 10 is considered to be appropriate in this case. Taking into account the relative sizes of the contributions due to internal and external events, the estimated overall change in the probability of core melt leading to Category 2 and 3 releases is likely to be subject to an error factor of about 7. More detailed analysis of the individual sequences might lead to some refinement of this estimate.

*Note: An error factor of 3, as used here, implies that the 95% confidence limits are believed to be a factor of 3 above and below the best estimate value.

There is also some additional uncertainty in the upward direction due to lack of completeness in the PRA arising from unquantifiable factors such as obscure common mode faults that have not been identified. However, as we are concerned here only with differential changes in the probability of core melt $p(m)$ and the probability of release $p(r)$ this source of uncertainty can be ignored in the first instance. A possible method of dealing with the unquantifiable components of $p(m)$ and q is discussed in Reference 31.

The conditional probability of releases to the environment in categories 2 and 3, given core melt can be denoted by q_2 and q_3 respectively. It is necessary to consider the uncertainties in q_2 and q_3 .

A value of $1.4E-2$ has been assumed for q_2 (see Table 7.2). It could conceivably be too low by a factor of 2 as judged by the testimony given by NRC at the Indian Point Hearing,³⁶ but this seems unlikely for ANO-1. It is considered unlikely that it is too high.

From Table 7.2 it can be seen that q_3 can be either 0.014 or 0.18, but from Table 7.4 it can be seen that for the important sequence it has the value 0.18. At the most it could therefore be too low by a factor of 5; it seems unlikely that it would be too large by a factor of more than 14.

Thus the estimated values for the conditional probabilities of releases in categories 2 and 3 due to the dominant sequences are likely to be conservative. Therefore the estimated probabilities of release due to these sequences are also likely to be conservative. The effect of these uncertainties is discussed further in Section 8.3.9 below.

8.2.6 Overall uncertainty in the value term $V(T)$

From the preceding discussion it can be seen that the main uncertainties in $V(T)$ and their causes are as follows:

- n_{eff} Factor of about 1.5 about a best estimate value of 13.2 years, arising from a possible change in NRC's choice of discount rate in the range 0 to 10%.
- C(D) The NRC's empirical guidance for estimating offsite costs on the basis of \$1000 per person rem of population dose is unlikely to lead to a lower estimate for ANO-1 than the more elaborate method of Aldrich and Strip^{33,34} which is discussed in Reference 30. However, conceivable changes of the "ground rules" assumed by Strip could increase his estimate by a factor of about 5; e.g., changes in decontamination cost alone could lead to an increase of about 60%; other factors, within the NRC's discretion, could lead to a 3-fold increase.

F(D) If decision making can be based on averaged weather conditions, the uncertainty in F(D) would be negligible compared with that in some of the other parameters. There is about a 10% chance that it could be at least a factor of 2 greater than the Central Estimate.

$\sum q_{jk} \Delta p(m)_j$ This term is likely to be subject to an error factor of about 7 in Δp_j and the further uncertainty in q_{jk} discussed in the previous section (i.e., upper and lower bounds which are factors of 5 above and 14 below the central value of 0.2).

From the summary it can be seen that, with the existing NRC "ground rules," the dominant cause of uncertainty is the random variation in the parameters which determine $\Delta p(m)_j$ and q_{jk} . This gives rise to an error factor of about 7 in $\Delta p(m)_j$.

Possible changes in the NRC ground rules for estimating population dose could lead to an increase in V(T) by a factor of as much as 5 (choice of a zero discount rate, combined with a change in interdiction dose from 25 rem to 5 rem over 30 years). An increase in the distance at which population dose is considered could also lead to a substantial increase in the dose.

The only likely decrease in V(T) arising from a change within NRC's discretion would be that associated with an increase in the discount rate (factor of 1.5 in going from 5% to 10% per year).

8.3 The Uncertainty in Combined Onsite and Offsite Costs

8.3.1 The Structure of the Onsite Cost Term

As indicated in the introduction to this section, it must be noted that in the discussion of the uncertainties, "avertable onsite costs" have been treated as "positive values," not as "negative impacts."

It is shown in Reference 30 that the onsite cost term, U(T), which is the present worth of the expectation of loss, can be represented by the following expression:

$$U(T) = n_{\text{eff}} p(m)(X+Y+Z) \quad (5)$$

where n_{eff} has the same meaning as in the previous section,

$p(m)$ is the overall probability of core melt per year

X is the differential cost of replacement power for the first 10 years after core melt

Y is the loss of capital investment remaining at the time of core melt

Z is the cost of on site cleanup after core melt

It is also shown in Reference 30 that the change in onsite cost, $\Delta U(T)$ due to an improvement which reduces $p(m)$ by $\Delta p(m)$, can be written

$$\Delta U(T) - n_{\text{eff}} \Delta p(m)(X+Y+Z) \quad (6)$$

The uncertainties in each of the terms on the right hand side of Eqn (6) are discussed in the following Sections

8.3.2 Uncertainty in n_{eff}

This is the same as in the case of offsite costs.

8.3.3 Uncertainty in $\Delta p(m)$

In a single accident sequence, j , the uncertainty in $\Delta p(m)_j$ may vary widely from that of $\Delta p(m)$. Detailed inspection of the individual sequences might show that the uncertainty in $\Delta p(m)$ is smaller than that of the corresponding term in Equation (3) but this extension of the analysis has not yet been carried out. Thus, for the present, it is assumed that the uncertainty in the quantifiable component $\Delta p(m)$ can be represented by an error factor of about 5.

8.3.4 Uncertainty in the Differential Cost of Replacement Power

As discussed in Reference 30, evaluation of the uncertainty in the differential cost of nuclear power presents difficulties of two different types: at the local level, changes in the ratio of supply to demand could have a significant effect on the cost of the replacement power from fossil fired stations (e.g., if the growth of demand were higher than expected, a higher proportion of the replacement power might have to be generated by oil fired plants or by older, less efficient, coal fired plants). At the national level escalation of either, or both, fossil or nuclear fuel costs at a rate different from the general inflation rate would lead to a change in the present worth of the replacement fuel cost, e.g., if the costs of both type of fuel were subject to an escalation rate, in real terms, of 5% per year, the present worth of the component of the onsite costs would be approximately twice its value in the absence of escalation.

In the case of the ANO-1 NPP, loss of one unit would represent a loss of only 18% of the electricity generating capacity of the coal fired units of the power pool (No. 20) which it serves³⁷ so that extensive use of expensive oil or gas fired units is presumed to be unlikely. Thus the uncertainty due to this cause is negligible.

In the event of an accident sequence leading to core melt in both units, the additional load demand on the coal fired units would represent 26% of the total capacity, leading to the use of a higher proportion of high fuel-cost plants. Thus, the cost of replacement power might increase by a factor of more than 2. However, the relative probabilities of these sequences is not high and replacement power is only 10% of the total onsite cost. Thus, the overall effect on the uncertainty in the onsite cost term is insignificant.

The escalation rates, in real terms of nuclear and fossil fuels in the No. 20 power pool, are the principal sources of uncertainty in the prediction of the replacement power costs. It seems likely that the escalation rates for both types of fuel would be at least 3% over the next 10 to 20 years.³⁸ A central value of 5% per year for each has therefore been used to estimate this component of the onsite costs; upper and lower limits of 7% and 3% per year should then provide reasonable upper and lower bounds for this component.

In this analysis it has been assumed that the replacement power costs in 1986 would be as estimated by VanKuiken in Reference 38. Escalation at 5% per year over the average remaining life of the plant gives a present worth for the avertable loss for 10 years replacement power of $\$2.0E10 \cdot p(m)$. If the costs of both types of fuel were to escalate at 3% and 7% per year, the present worth would become $\$1.6E10 \cdot p(m)$ and $\$2.5E10 \cdot p(m)$, respectively, i.e., the uncertainty could be represented by a factor of 1.2 about the central value.

Other cases can be postulated in which the two fuels have different escalation rates; in the extreme cases where these are 3% and 7%, the present worth becomes $\$1.6E10 \cdot p(m)$ for escalation rates of 7% per year for nuclear and 3% per year for fossil fuel. In the converse case the present worth becomes $\$2.9E10 \cdot p(m)$. Thus, in the case where nuclear fuel costs are assumed to increase much more rapidly than fossil fuel costs, the present worth of the differential fuel costs would be 80% of the central value. In the converse situation, however, the change would be smaller, increasing by a factor of about 1.5 over the central value. For reasons which are discussed in Reference 30, it is considered unlikely that either of these situations would arise during the design lifetime of the ANO-1 NPP, particularly the case of more rapid escalation of nuclear costs.

Thus, it is considered that the overall uncertainty in the value of the replacement power cost can be represented by a factor of 1.5 about the central case in which the present nuclear and fossil fuel costs estimated in Reference 37 both escalate by 5% per year in real terms. It should be noted that in this central case the replacement power cost is twice that which would be estimated on the assumption that neither fuel cost escalated.

Inclusion in the value-impact analysis of the possibility of accidents leading to core melt in both units would not lead to a significant increase in the uncertainty in the onsite cost term.

8.3.5 Uncertainty in the Evaluation of the Loss of the Utility's Investment

In Reference 30 a method for evaluating the utility's investment at the time core melt occurs is described. The method is designed to express the value of the investment in 1985 dollars, allowing for depreciation proportional to the number of years of commercial operation preceding core melt.

Given that the method described is acceptable, the monetary value attributed to the investment should not be subject to any significant uncertainty and has not been investigated further.

8.3.6 Uncertainty in the Cost of Cleanup After Core Melt

As indicated in Section 6 of this report, the "rule of thumb" figure of $\$1.2 \times 10^9$ proposed by NRC³² has been used in the value-impact analysis for the ANO-1 NPP. There is a general consensus of technological opinion that the cost of cleanup after core melt would be in the range $\$1 \times 10^9$ to $\$2 \times 10^9$. Thus, the "rule of thumb" figure is unlikely to be in error by a factor of more than 2.

8.3.7 Uncertainty Due to the Omission of Possible Components of the Onsite Costs

In Reference 30 a number of other possible contributions to the onsite cost term are discussed. Inclusion of these items could increase the total onsite costs due to the 3 components discussed above (i.e., the terms X, Y, and Z in Equation 5) by a factor of about 2. The decision as to whether or not these additional costs should be included rests with NRC.

8.3.8 Summary of the Uncertainty in the Onsite Cost Term

By summing the lower and the upper bound values for the 3 onsite monetary terms (X, Y, and Z of Equation 5) overall limits to the monetary terms are found to be $3.2E+10$ and $5.9E+10$. Thus a reasonable estimate for the central value is

4.5E10, with upper and lower bounds a factor 1.4 above and below the central value.

The uncertainties in the various components of the onsite cost terms are summarized in Table 8.2. It can be seen that the overall uncertainty is dominated by that of Δp_m ; the next most important contribution arises from items where NRC decisions are required; the uncertainty in the monetary terms (X, Y, and Z) is relatively unimportant.

8.3.9 The Uncertainty in the Combined Onsite and Offsite Costs

8.3.9.1 The Relative Sizes of the Offsite and Onsite Costs

In other chapters of this report central values are given for the present worths of the maximum expectations of loss attributable to avertable offsite and onsite costs associated with possible failures of DHR related systems of the ANO-1 NPP in its present form. These central values are as follows:

Offsite costs	1.4E+5
Onsite costs	5.5E+6

Thus it can be seen that the onsite costs are more than 33 times as large as the offsite costs. It is to be expected that the ratio of the two types of costs would be similar after each of the various sets of modifications which have been examined.

In accident sequences that could lead to core melt in both units the onsite costs would be doubled but, as noted in Section 8.2.4, above, population dose would be unlikely to increase by more than 10%. Thus the ratio of offsite to onsite costs would be reduced by a factor of about 1.9.

8.3.9.2 The Uncertainty in the Combined Offsite and Onsite Costs

Since the overall uncertainty in the onsite terms is substantially less than that in the offsite terms and the size of the former is about 38 greater, the overall uncertainty in the total averted costs associated with a particular set of modifications will be nearer to that of the onsite term. Using the results obtained in other Sections, this can be demonstrated quantitatively for 2 of the Alternatives (the other 3 yield similar results), as shown in Table 8.3. It should be noted that Table 8.3 does NOT include the effects of possible changes which are at the discretion of NRC.

In completing Table 8.3 the upper and lower bounds for offsite and onsite costs have been estimated using the uncertainty factor in Tables 8.1 and 8.2 for each parameter (other than those where a decision by NRC is concerned) combined together, i.e., the factors (x and +) shown in the third columns of

Tables 8-1 and 8-2, as representing the "Likely Range" in the various parameters for which probability distributions are not known, have been applied to obtain bounding values for $\Delta U(T)$ and $\Delta V(T)$.

The sensitivity of the results to changes in the assumptions made about NRC's decisions concerning certain parameters can be seen by examining their effects on the central values in Table 8.3, viz:

- i) Increasing the averted offsite costs by a factor of 5, (e.g., to take account of possible changes in the NRC ground miles for estimating population dose out to 50 miles) would only increase the combined averted cost by a factor of 1.1.
- ii) Increasing the averted onsite costs by a factor of 2, to take account of other possible causes of loss, would increase the combined averted cost by 98%, essentially doubling it.
- iii) A change in discount rate from 5% to 10% per year would reduce all averted costs by a factor of 1.5.
- iv) A change to zero discount rate for averted offsite costs only would increase these by a factor of 1.4 but would only increase the combined averted costs by a factor of about 1.03.

A change in method used to estimate offsite and onsite avertable costs, from NRC's empirical rules to the less empirical method used by Strip in Reference 33 (i.e., basically the same as in the CRAC2 code, for offsite costs), would not have a major effect on the overall uncertainty.

8.4 Effects of Uncertainties in the Impact Term

The principal sources of uncertainty in the impact term are:

- i) Uncertainties in the costs of the various "Alternatives."
- ii) The effects of the various "Alternatives" on the availability of the NPP during the remainder of its service life.

As discussed in Reference 30, the uncertainty in the cost estimates for the "Alternatives" depends largely on the degree of detail to which the design of the alternative is carried.

It is expected that only Alternative 2 would have any significant effect on availability. As it provides another, independent, system for removing decay heat, it is possible that it could improve availability by avoiding the need for a shutdown, in order to comply with the Technical Specifications for the NPP, following the loss of one of the other systems for decay heat removal.

In the case of ANO-1 a saving of only one day's outage per unit per year, on average, over the remaining life of the station, would have a present worth of \$6.9 E+6. The present worths of other savings would be pro rata.

In principle, it should be possible to evaluate the number of days that could be saved at ANO-1 with relatively little error, since plant specific failure rates should be obtainable. However this has not yet been attempted. Consequently it is not possible at this stage to discuss the effect of this part of the impact term on the overall uncertainty in the analysis.

8.5 The Overall Uncertainty in the Value-Impact Analysis

For the reasons discussed in Reference 30, when the uncertainty in either the value terms or the impact terms (or both) is large, it is more useful to express the results of the analysis in terms of a non-dimensional ratio rather than as a "Net Benefit" in dollars. In this case we have an uncertainty in the Value term which can be represented by upper and lower bounds that are a factor of 9 on either side of the central value whereas in the Impact term the corresponding bounds would be a factor of less than 2 about the central value. Whatever the forms of the frequency distribution for the Value and Impact terms, it seems highly likely the upper and lower bounds on their ratio would be at a factor of about 9 on either side of the mean. For example, for illustrative purposes, if it were assumed that each term could be represented by log normal functions with Error Factors of 9 and 2 respectively, the Error Factor of their ratio would be 10.

Table 8.1

ANO-1 Nuclear Power Plant--Summary of the Variation
in the "Value" Term, if Based on Offsite Costs Only,
With the Principal Parameters Subject to Uncertainties

Parameter	Central Value	Likely Range	Effect on Offsite Cost	Nature of the Uncertainty
Discount Rate	5% p.a.	0% x 1.4 10% ÷ 1.5	Linear	NRC decision
Change in Core Melt Probability Δp_m	Variable	x 7 ÷	Linear	Statistical
Probability of containment failure q_i (Dominant sequences)	0.2	x 5 ÷ 14	Linear	Partly statistical, partly lack of knowledge
Change in probability of large release Δp_{ri}	Variable	x 35 ÷ 98	Linear	As q_i
Release Fractions (Dominant Sequences)	0.3 RSS	x 3 ÷	x 1.1 ÷	Lack of knowledge
Possible changes in the ground rules for estimating population dose to 50 miles	NRC Discretion	x 5 ÷ 1	Linear	NRC decision

Table 8.2

ANO-1 Nuclear Power Plant--Summary of the Variation
in the "Value" Term, if Based on Onsite Costs Only, With
the Principal Parameters Subject to Uncertainties

Parameter	Central Value	Likely Range	Effect on Onsite Cost	Nature of the Uncertainty
Discount Rate	5% p.a.	0% x 1.4 10% ÷ 1.5	Linear	NRC decision
Change in Core Melt Probability Δp_m	Variable	x 7 ÷	Linear	Statistical
Change in the monetary terms in the Value analysis	See Value analysis	x 1.3 ÷	Linear	Lack of knowledge
Possible additional contributions to onsite costs	See Value analysis	x 2.0 ÷ 1.0	Linear	NRC decisions

Table 8.3 ANO-1 Nuclear Power Plant

Uncertainty in Combined Offsite and Onsite Costs (Present Worths
in Dollars) due to Factors NOT Dependent on NRC Decisions

Alternative No.	Averted Offsite Cost			Averted Onsite Cost			Combined Averted Cost			Uncertainty Factor in Combined On- and Offsite Costs
	Central Value	Lower Bound	Upper Bound	Central Value	Lower Bound	Upper Bound	Central Value	Lower Bound	Upper Bound	
		+ 98	x 35		+ 9.1	x 9.1				
1	1×10^5	1.02×10^3	3.5×10^6	3.93×10^6	4.3×10^5	3.58×10^7	4.03×10^6	4.3×10^5	3.93×10^7	x 9.8 + 9.4
2	1.31×10^5	1.33×10^3	4.6×10^6	5.07×10^6	5.6×10^5	4.6×10^7	5.2×10^6	5.61×10^5	5.06×10^7	x 9.7 + 9.3

9.0 INTEGRATED VALUE-IMPACT ANALYSIS

In this section the quantitative value analysis (Section 7.0) and the impact analysis (Section 6.0) are combined to form an integrated value-impact analysis. Unquantifiables are not treated in this section but are addressed in various other sections. The goal of this assessment is to provide measures that can be used by the NRC to make decisions on the adequacy of decay heat removal requirements (USI A-45). The methodology for the value-impact analysis is presented in Appendix L. In order to implement the value-impact analysis and illustrate the steps performed, the methodology will be reviewed briefly prior to summarizing the ANO-1 results.

9.1 Methodology

9.1.1 Value and Impact Analysis Variables

Each of the variables to be used as input in the V-I analysis is defined in Table 9.1 and characterized in several ways. First, the costs and values may be incurred one time or on an annual basis. All recurring costs or costs which might occur at any time during the remaining plant lifetime must be present valued. The present worth factor for ANO-1 based upon 22 years remaining life is 13.2 at a 5% discount rate. Second, there are positive and negative values and impacts. For example, in the case of I_4 and V_4 the modifications could result in an increased or decreased value or impact. The modifications could require additional maintenance resulting in increased radiation exposure and/or longer outage periods and thus more replacement power costs. On the other hand, the modifications could make maintenance easier, thus reducing radiation exposure and/or replacement power costs. Third, costs or doses result from either the proposed modifications or an accident. Fourth, each variable may affect the utility, and the NRC, or the public. Last, the information for each variable comes from the architect engineer, the value analysis, or other, previously documented, analyses.

In addition to the value and impact variables, the change (reduction) in core melt probability per reactor year (Δp_m) for each alternative from the base case core melt probability per reactor year (i.e., without any modifications) is used.

$$\Delta p_m(j) = p_m - p_m(j), \text{ central value}$$

where p_m = base case core melt probability, and

$$p_m(j) = \text{core melt probability of the } j\text{th alternative.}$$

These values are given in Section 7.1.1. The upper (U) or lower (L) bounds for the change in core melt probability are determined as follows:

Table 9.1 Value and Impact Analysis Input Variables

<u>Symbol</u>	<u>Description</u>		<u>Results from</u>	<u>Affects the</u>	<u>Source of Information</u>
I ₁	Engineering and Installation Cost - one time	Positive Impact	Modifications	Utility	Architect Engineer
I ₂	Operations and Maintenance Costs/year - present worth	Positive Impact	Modifications	Utility	Architect Engineer
I ₃	Installation Replacement Power Costs - one time	Positive Impact	Modifications	Public	Architect Engineer
I ₄	In-service Replacement Power Costs - per year - present worth	Positive or Negative Impact	Modifications	Public	Not available - but probably negligible
I ₅	Avertable Onsite Costs - one time - present worth	Negative Impact	Accident	Utility	Based on previous analyses
I ₅₁	Replacement Power Costs				
I ₅₂	Loss of Investment Costs				
I ₅₃	Site Cleanup Costs				
I ₆	Other Costs - one time	Positive Impact	Modifications	Utility & NRC	Not covered in this program
V ₁	Averted Onsite Dose Over Plant Lifetime	Positive Value	Accident	Utility	Based on previous analyses
V ₁ '	Present Worth of Averted Onsite Dose @ \$1000/p-rem	Positive Value	Accident	Utility	Based on previous Analyses

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Table 9.1 Value and Impact Analysis Input Variables (continued)

<u>Symbol</u>	<u>Description</u>		<u>Results from</u>	<u>Affects the</u>	<u>Source of Information</u>
V ₂	Averted Offsite Dose Over Plant Lifetime	Positive Value	Accident	Public	From value analyses
V ₂ '	Present Worth of Averted Offsite Dose @ \$1000/p-rem	Positive Value	Accident	Public	From value analyses
V ₃	Installation Dose - one time	Negative Value	Modifications	Utility	Architect Engineer
V ₃ '	Present Worth of Installation Dose @ \$1000/p-rem	Negative Value	Modifications	Utility	Architect Engineer
V ₄	In-service Operational Dose over Plant Lifetime	Positive or Negative Value	Modifications	Utility	Not available
V ₄ '	Present Worth of Occupational Dose @ \$1000/p-rem	Positive or Negative Value	Modifications	Utility	Not available

$$\text{Upper bound } \Delta p_m(j) = \Delta p_m(j) \times 5$$

$$\text{Lower bound } \Delta p_m(j) = \Delta p_m(j) \div 5$$

The factor of 5 uncertainty in Δp_m was estimated by assuming a factor of 3 uncertainty for the internal event sequences and a factor of 10 for the special emergency sequences as shown in Table 9.2. When these factors are applied to the central value or best estimate for each initiator, i.e., internal, seismic, spray, etc., and summed, the overall core melt bounds for the initiators considered in this program are $3.81\text{E-}5$ for the lower bound and $1.18\text{E-}3$ for the upper bound. While these bounds are not exactly symmetrical about the central value, the range corresponds to approximately a factor of 5.

9.1.2 Value Impact Analysis Measures

Impact Measures - Impacts I_2 and I_4 (refer to Table 9.1) must be multiplied by the present worth factor, 13.2 for ANO-1 assuming 5% discount rate. The present worth factor accounts for the reduced worth of the payments made or incurred at some future date. Therefore:

$$\text{Total Positive Impact of the alternative, } TI(j) = I_1(j) + 13.2 I_2(j) + I_3(j) + 13.2 I_4(j).$$

This is the total of all the positive impacts. The negative impacts sum to the,

$$\text{Total avertable cost, } I'_5(j) = I'_{51}(j) + I'_{52}(j) + I'_{53}(j).$$

where

$$I'_{51}(j) = \Delta p_m(j) \times I_{51}(j)$$

$$I'_{52}(j) = \Delta p_m(j) \times I_{52}(j)$$

$$I'_{53}(j) = \Delta p_m(j) \times I_{53}(j)$$

Upper and lower bounds are introduced at this point in the impact analysis summary using the error factor of 5 discussed previously. That is,

$$\text{Lower bound (L) } I'_5(j) = I'_5(j) \div 5 = I_5(j) \times \Delta p_m(j) \div 5.$$

$$\text{Central value (C) } I'_5(j) = I_5(j) \times \Delta p_m(j), \text{ and}$$

$$\text{Upper bound (U) } I'_5(j) = I'_5(j) \times 5 = I_5(j) \times \Delta p_m(j) \times 5.$$

The same is true for $I'_{51}(j)$, $I'_{52}(j)$, and $I'_{53}(j)$. Therefore, for the j th alternative,

Table 9.2 ANO-1 Value-Impact Analysis Error Factor

<u>Analysis</u>	<u>Core Melt Probability</u>		
	<u>Lower Bound</u>	<u>Central</u>	<u>Upper Bound</u>
Internal	2.9E-5	8.8E-5	2.6E-4
Seismic	7.3E-6	7.3E-5	7.3E-4
Fire	5.8E-7	5.8E-6	5.8E-5
Wind & Missiles	5.3E-7	5.3E-6	5.3E-5
External Flood	7.2E-7	7.2E-6	7.2E-5
Lightning	<u>1.8E-8</u>	<u>1.8E-7</u>	<u>1.8E-6</u>
	3.81E-5	1.79E-4	1.18E-3

Net Impact = Total Impact - Total Avertable Impacts or

$$NI(j) = TI(j) - I'_5(j)$$

The discounted values for the negative impacts due to avertable onsite costs, conditional upon an accident, are from Appendix L are:

$$I_{51} = \$11.9E+9,$$

$$I_{52} = \$3.05E+9, \text{ and}$$

$$I_{53} = \$1.58E+10 \text{ in 1985/1986 dollars.}$$

Value Measures - The averted onsite dose (V_1) for each alternative is estimated from onsite dose received during an accident. For purposes of this analysis, this onsite dose (50,600 person-rem) is assumed to be the same for any core melt accident as discussed in Appendix L. The onsite dose (50,600 person-rem) is multiplied by the Δp_m for the j th alternative and by the number of years of operation remaining. Thus,

$$V_1(j) = (50600) \times \Delta p_m(j) \times 22$$

The present worth, $V_1'(j)$, of the above avertable onsite dose, valued at \$1000 per person-rem is:

$$V_1'(j) = (50600) \times \$1000 \times \Delta p_m(j) \times 13.2$$

The offsite averted dose (V_2) for each alternative is estimated from the avertable offsite dose per reactor year (given for each alternative in Section 7.0) multiplied by the remaining years of plant operation. Thus

$$V_2(j) = (\text{Averted doses from Table 7.8}) \times 22$$

In this case the Δp_m is not required because the core melt probability is inherent in the calculations in Section 7.

The present worth $V_2'(j)$ of the avertable offsite dose valued at \$1000 per person-rem is:

$$V_2'(j) = (\text{Averted doses from Table 7.8}) \times \$1000 \times 13.2.$$

The totals of the positive values (onsite + offsite) for averted dose and costs are:

$$V_{12}(j) = V_1(j) + V_2(j)$$

$$V_{12}'(j) = V_1'(j) + V_2'(j)$$

The ratio of the averted offsite dose (V_2) from Table 7.8 and the base case dose from Table 7.7 is

Averted dose ratio, offsite $\equiv$

$$R_o = \frac{V_2(ji)}{\text{Base Case Dose } (i) \times 22}$$

where i represents the lower, central, and upper bound values.

Similarly, the ratio of the total averted dose (V_{12}) and the total base case dose is

Averted dose ratio, onsite and offsite $\equiv$

$$ADR_n = \frac{V_{12}(ji)}{\text{Base Case Dose } (i) \times 22 + V_1(j)}$$

The negative values considered for the j th alternative included the dose received during installation (a one time dose), $V_3(j)$, and the in-service occupational dose received over the remaining plant life time, $V_4(j)$. These doses can be presented valued at \$1000 per person-rem in a manner analogous to that used for the positive values, thus,

$$V_3'(j) = V_3(j) \times \$1000$$

$$V_4'(j) = (V_4(j)/22) \times \$1000 \times 13.2$$

$V_3'(j)$ does not include the present worth factor (13.2) because it is a one time dose, whereas the dose associated with operations, $V_4(j)$ is recurring.

The net averted dose, $NV(j)$, and net averted costs, $NV'(j)$ associated with each alternative can be calculated by subtracting the negative values from the positive. Thus

$$\text{Net averted dose} = NV(j) = V_1(j) + V_2(j) - V_3(j) - V_4(j)$$

$$\begin{aligned} \text{Net present worth} &= NV'(j) = V_1'(j) + V_2'(j) - V_3'(j) - V_4(j) \\ \text{of averted costs} & \\ \text{at } \$1000/p\text{-rem} & \end{aligned}$$

Value-Impact Measures - The value-impact measures can be constructed from the variables defined above. Each of the value-impact measures is calculated with the central cost for the total impact (TI) and the central cost for the net impact (NI) as opposed to using the upper and lower bounds for NI.

The first measure considered is the a ratio of averted costs to impacts. This value-impact ratio (VIR) is calculated twice.

One ratio considers only the averted offsite costs and the total impact. Thus for the jth alternative

$$VIR_0 \text{ (offsite)} = V_2'(j)/TI(j) \quad .$$

The second ratio is the net value-impact ratio which accounted for the net averted costs and the net impacts.

$$VIR_n \text{ (offsite and onsite)} = NV'(j)/NI(j) \quad .$$

Similarly there are two net benefit values. The first again considers only averted offsite costs, while the second uses net averted costs and impacts. So one has,

$$NBV_0 \text{ (offsite)} = V_2'(j) - TI(j)$$

$$NBV_n \text{ (offsite and onsite)} = NV'(j) - NI(j)$$

The final value-impact measure presented is the estimated cost in dollars per person-rem of Dose Averted if the alternative is implemented. Again there are two values, one considering only the offsite averted costs, the second the net costs.

$$DPR_0 \text{ (offsite)} = TI(j)/V_2(j)$$

$$DPR_n = \text{(offsite and onsite)} = NI(j)/NV(j) \quad .$$

A more complete discussion of the reasons for selecting these measures is provided in Appendix L.

9.2 Results - The values, impacts, and value impact measures defined in the previous section were calculated for the ANO-1 plant using the results of the internal analysis, special emergency analyses, and the impact analysis. These results are tabulated in Tables 9.3, 9.4, and 9.5. It is important to note that the offsite population dose is an integrated dose out to a radius of 50 miles from the site and the conversion from dose to cost is at \$1000 per person rem. All present value estimates are based upon a 5% discount rate. Each of these tables gives the upper (U), central (C), and lower (L) bounds when applicable.

The symbols for the values, impacts, and measures are given at the heading of each column to avoid any possible confusion about the entries in the table or how the numbers were determined.

Table 9.3 summarizes the impacts for ANO-1 by alternative. The four positive impacts are individually tabulated and then totaled to obtain the Total Impact (TI) of modifications associated with each alternative. One time costs of installing the modifications and replacement power during the installation are already in present worth dollars. The in service

(Text continued on page 9-13.)

TABLE 9.3 ANO-1 - SUMMARY OF IMPACTS (Based Upon 5% Discount)

Alternative No.	Change in Core Melt Probability [Central Value]	POSITIVE IMPACTS ASSOCIATED WITH MODIFICATIONS (Present Worths)					NEGATIVE IMPACTS DUE TO AVERTABLE ONSITE COSTS (Present Worths)				NET IMPACT (\$ x10 ⁻⁶)
		Utility Costs		Replacement Power Costs		TOTAL POSITIVE IMPACT (\$ x10 ⁻⁶)	Replace- ment Power Costs (\$ x10 ⁻⁶)	Loss of Invest- ment Costs (\$ x10 ⁻⁶)	Site Cleanup Costs (\$ x 10 ⁻⁶)	Total Avert- able Costs (\$ x10 ⁻⁶)	
		Installation and Engineering Costs (\$ x10 ⁻⁶)	Operations and Maintenance Costs (PW) (\$ x10 ⁻⁶)	Installation (\$ x10 ⁻⁶)	In Service (PW) (\$ x10 ⁻⁶)						
	Δp_m	I_1	13.2 I_2	I_3	13.2 I_4	TI	[U, C, & L Values]				NI
1	1.28E-4	14.298	1.056	0.0	Not Available Probably Negligible	15.354	L 0.30 C 1.52 U 7.62	0.08 0.39 1.95	0.40 2.02 10.11	0.78 3.93 19.68	14.574 11.424 - 4.326
2	1.65E-4	54.697	4.660	0.0	"	59.357	L 0.39 C 1.96 U 9.82	0.10 0.50 2.52	0.52 2.61 13.04	1.01 5.07 25.38	58.347 54.287 33.977

$$TI = I_1 + 13.2 I_2 + I_3 + 13.2 I_4$$

$$I_5' = I_{51}' + I_{52}' + I_{53}'$$

$$NI = TI - I_5'$$

Note: L, C, and U represent the lower, central, and upper bound estimate for core melt probability

TABLE 9.4-a ANO-1 - SUMMARY OF VALUES (Based on Population Dose to 50 Miles, 5% Discount Rate)

Alternative No.	Change in Core Melt Probability [Central Value]	POSITIVE VALUES									
		Onsite		Offsite			Total				
		Averted Dose (p-rem)	Present Worth of Averted Dose @ \$1000/p-rem (\$ x10 ⁻⁶)	Averted Dose (p-rem)	Averted Dose ÷ Base Case Dose	Present Worth of Averted Dose @ \$1000/p-rem (\$ x10 ⁻⁶)	Averted Dose (p-rem)	Averted Dose ÷ Base Case Dose	Present Worth of Averted Dose @ \$1000/p-rem (\$ x10 ⁻⁶)		
Δp_m		V_1	V_1'	V_2	{U, C, & L Values}		{U, C, & L Values}		V_{12}	ADR_n	V_{12}'
1	1.28E-4	142	0.085	L 103	.68	0.062	245	.83	0.147		
				C 167	.70	0.100	309	.81	0.185		
				U 288	.70	0.173	430	.77	0.258		
2	1.65E-4	184	0.110	L 139	.92	0.083	323	.96	0.193		
				C 218	.91	0.131	402	.95	0.241		
				U 378	.91	0.227	562	.94	0.337		

$$\begin{aligned}
 V_1 &= 50600 \times \Delta p_m \times 22 \\
 V_1' &= 50600 \times \Delta p_m \times \$1000 \times 13.2 \\
 V_2 &= \text{Averted Dose} \times 22 \\
 V_2' &= \text{Averted Dose} \times \$1000 \times 13.2 \\
 ADR_o &= V_2 \div (\text{Basecase Dose} \times 22) \\
 V_{12} &= V_1 + V_2 \\
 ADR_n &= V_{12} \div (\text{Basecase Dose} \times 22 + V_1) \\
 V_{12}' &= V_1' + V_2'
 \end{aligned}$$

Note: L, C, and U stand for lower, central, and upper bound source term estimate

TABLE 9.4-b ANO-1 - SUMMARY OF VALUES (Based on Population Dose to 50 Miles, 5% Discount Rate)

Alternative No.	Change in Core Melt Probability [Central Value]	NEGATIVE VALUES						Net Value	
		Installation		Operation		Total			
		Installation Dose (p-rem)	Present Worth of Installation Dose @ \$1000/p-rem	In-Service Operational Dose (p-rem)	Present Worth of In-Service Oper. Dose @ \$1000/p-rem	Installation and Operational Dose (\$ x10 ⁻⁶)	Present Worth of Install. & Oper. Dose @ \$1000/p-rem (\$ x10 ⁻⁶)	Averted Dose (p-rem)	Present Worth of Averted Dose @ \$1000/p-rem (\$ x10 ⁻⁶)
	Δp_m	V_3	V_3'	V_4	V_4'	$V_3 + V_4$	$V_3' + V_4'$	[L, C, & U Values]	
								NV	NV'
1	1.28E-4	0	0	Negligible		0	0	L 245 C 309 U 430	0.147 0.185 0.258
2	1.65E-4	20	.020	Negligible		20	.020	L 303 C 382 U 542	0.173 0.221 0.317

$$V_3' = V_3 \times \$1000$$

$$V_4' = (V_4 \div 22) \times \$1000 \times 13.2$$

$$NV = V_1 + V_2 - V_3 - V_4$$

$$NV' = V_1' + V_2' - V_3' - V_4'$$

Note: L, C, and U stand for lower, central, and upper bound source term estimate

TABLE 9.5 AND-1 - SUMMARY OF VALUE-IMPACT ANALYSIS (Based on Population Dose to 50 Miles; 5% Discount Rate)

V-I ANALYSIS BASED ON OFFSITE COSTS														V-I ANALYSIS BASED ON OFFSITE AND ONSITE COSTS													
Alter- native No.	Change in Core Melt Probability [Central Value]	Offsite Averted Dose [p-rem]	Total Averted Dose 1 Base Case Dose	Total Impact [Central Cost]	Present Worth of Averted Dose @ \$1000/ p-rem [U,C,&L Values]		Measures of V-I			Net Impact [Central Cost]	Present Worth of Averted Dose @ \$1000/ p-rem [U,C,&L Values]	Measures of V-I															
					V-I Ratio	Net Benefit	Dollars per p-rem	V-I Ratio	Net Benefit			Dollars per p-rem															
1	1.28E-4	L 103	.83			0.062	4.03E-3	-15.292	14.9E+4		0.147	1.29E-2	-11.78	4.7E+4													
		C 167	.81	15.354	0.100	6.51E-3	-15.254	9.2E+4	11.424	0.185	1.62E-2	-11.24	3.7E+4														
		U 288	.77		0.173	1.13E-2	-15.181	5.3E+4		0.258	2.26E-2	-11.17	2.7E+4														
2	1.65E-4	L 139	.96			0.083	1.40E-3	-59.274	4.3E+5		0.173	3.19E-3	-54.11	1.8E+5													
		C 218	.95	59.357	0.131	2.21E-3	-59.226	2.7E+5	54.287	0.221	4.07E-3	-54.07	1.4E+5														
		U 378	.94		0.227	3.82E-3	-59.130	1.6E+5		0.317	5.84E-3	-53.97	1.0E+5														

$$VIR_o = V_2' + TI$$

$$NBV_o = V_2' - TI$$

$$DPR_o = TI + V_2$$

$$VIR_n = NV' + NI$$

$$NBV_n = NV' - NI$$

$$DPR_n = NI + NV$$

Note: L, C, and U stand for lower, central, and upper bound source term estimate

operations and maintenance costs and replacement power costs must, however, be multiplied by 13.2 to account for the present worth of these impacts. The installation of modifications at ANO-1 can be accomplished during normal outages so that there are no replacement power costs. Although replacement power costs due to in service maintenance were not specifically estimated, our judgment is that these costs will have a negligible impact. It is obvious from the table that alternative 2 will cost substantially more than alternative 1 due to the extent of the modifications.

The negative impacts result from the averted onsite costs attributed to a potential accident. The costs are, of course, probabilistic. Thus the potential costs for I_{51} , I_{52} , and I_{53} in 1985/1986 dollars must all be multiplied by the lower (L), central (C), and upper (U) bound values for Δp_m . The net impact (NI) is the positive impacts minus the negative impacts. The lower the net impact the more favorable the alternative appears. In fact, in one case (alternatives 1 upper bound) the net impact is negative which means that the averted onsite costs due to a possible accident at the upper bound change in core melt probability are greater than the cost of installing the modifications for those alternatives.

Table 9.4a presents the positive values for each alternative. These are onsite averted dose (V_1) and offsite averted dose (V_2) due to a potential accident. These are both probabilistic in nature, however, the calculation of offsite averted dose (V_2) does not explicitly include Δp_m since it is implicitly included in the analysis to obtain V_2 . Upper and lower bounds are given for V_2 as derived from the risk analysis in Section 7 and based on the source term bounds assumed in Section 7. Both averted doses must include a factor to account for the remaining plant life which is 22 years in 1985.

The total averted dose (V_{12}) is also given as are the present worth dollar values. Each of the dollar values is based on \$1000/p-rem and a 13.2 percent worth factor as described in Appendix L. The averted doses increases from alternative 1 to alternative 2 due to the increase in Δp_m from alternative 1 to alternative 2.

Table 9.4b summarizes the negative values for each alternative. The installation dose (V_3) results from radiation exposure to contractor personnel during installation of the modifications for any particular alternative. In-service occupational dose is considered to be negligible at ANO-1 for the alternatives proposed here. In each case the doses are converted to dollars by \$1000/p-rem. V_3 is already a present worth but V_4 requires a 13.2 present worth factor.

The net value is the positive values V_{12} or V_{12}' minus the negative values for $V_3 + V_4$ and $V_3' + V_4'$ respectively. Upper and lower bounds are given for NV and NV' which result from the bounds from the positive values only.

Table 9.5 is a summary of the Value-Impact analysis and as such several measures are repeated from Tables 9.3, 9.4a, and 9.4b. These repeated measures are TI, NI, V_2 , V_2' , ADR_n , and NV'. The value-impact measures derived from these measures are the Value-Impact Ratio (VIR), the Net Benefit Value (NBV), and the Dollars per person-rem (DPR) based on offsite costs alone (subscript o) and based on offsite and onsite costs combined (subscript n). Upper and lower bounds are shown where applicable. Recall that the V_2 , V_2' , ADR_n , and NV' value measure bounds are derived from the source term bounds discussed in Section 7 (that is approximately a ± 3 error factor). The impact measures TI and NI are central values even though upper and lower bounds are shown for NI in Table 9.3 to illustrate the range these onsite costs could have.

The results given in Table 9.5 show moderate differences in value-impact measures (VIR, NBV, and DPR) between an analysis based on offsite costs alone and an analysis based on offsite and onsite costs combined, the later being more favorable.

Table 9.6 is a further condensation of the results which show eight measures extracted from Table 9.5 for the central value only. While Δp_m and averted offsite dose (V_2) improve from alternative 1 to alternative 2, the reverse is true of the three value-impact measures (VIR, NBV, and DPR). In particular the dollars per person-rem are better for alternatives 1 and 2 if both onsite and offsite costs are considered. However, this part of the analysis does not consider any of the unquantifiables, thus judgments are deferred to Section 11 of this report where the quantifiable results found in this section are brought together with the unquantifiables.

TABLE 9.6 ANO-1 - SUMMARY OF VALUE-IMPACT MEASURES (Central Value)

Alternative No.	Change in Core Melt Probability	Offsite Averted Dose (p-rem)	Total Averted Dose Ratio	V-I Analysis of Offsite Costs			V-I Analysis Based on Offsite and Onsite Costs		
				V-I Ratio	Net Benefit (\$ x 10 ⁻⁶)	Dollars per p-rem	V-I Ratio	Net Benefit (\$ x 10 ⁻⁶)	Dollars per p-rem
	Δp_m	V_2	ADR_n	VIR_o	NBV_o	DPR_o	VIR_n	NBV_n	DPR_n
1	1.28E-4	167	0.81	6.51E-3	-15.25	9.2E+4	1.62E-2	-11.24	3.7E+4
2	1.65E-4	218	0.95	2.21E-3	-59.23	2.7E+5	4.07E-3	-54.07	1.4E+5

10.0 SPECIAL ISSUES RELATED TO DECAY HEAT REMOVAL

There has been considerable interest for many years in the use of feed and bleed (F&B) on the primary system as an alternative, essentially last resort, measure to remove decay heat from the reactor core. In addition, during the course of this investigation and other studies related to steam generator tube ruptures, questions have arisen related to the use of rapid secondary side blow-down to remove decay heat and depressurize the primary sufficiently to permit residual heat removal system cut-in. These two issues are discussed here, including some results from thermal-hydraulic studies on both subjects conducted on behalf of TAP A-45 by the Los Alamos National Laboratory.

10.1 Feed and Bleed

Feed and bleed is used more or less generically to describe two similar though somewhat different processes. In some plants, primarily those designed by Babcock and Wilcox, the high pressure injection (HPI) pumps have sufficient head and capacity to lift primary safety relief valves (SRVs) in addition to the power operated relief valves (PORVs) and provide sufficient flow to remove decay heat. Therefore, they can literally feed and bleed with liquid only (no void in the primary). In other plants the HPI pumps do not have sufficient head to lift the SRVs and so the PORVs must be opened first to reduce primary pressure to the point where HPI can provide adequate flow. In these plants the feed is liquid, but the bleed is usually steam or a two phase mixture, i.e., there will be some void in the primary. Detailed thermal-hydraulic analysis of feed and bleed phenomena and capabilities are reported elsewhere.^{13,14,15,16} In general, those studies support the concept of feed and bleed but they do point out that timing is a critical parameter in establishing whether or not primary feed and bleed can successfully remove decay heat.

10.1.1 Value of Feed and Bleed

In the case of ANO-1 Nuclear Plant, discussions with the plant staff revealed that feed and bleed procedures already exist. Therefore, in constructing the event trees (Appendix B) credit was given for the capability in the base case calculations. As reported in Appendix B, the ability to feed and bleed can influence twenty-three potential accident sequences. If the plant cannot feed and bleed these twenty-three sequences reduce to thirteen sequences. The specifics of the sequences replaced are discussed in Appendix B. It is sufficient for this discussion to note that the combined internal core melt probabilities considering the affected sequences after recovery analysis are (see Table 10.1):

P(cm) (internal events) without feed and bleed	1.23E-3
P(cm) (internal events) with feed and bleed	8.76E-5

Table 10.1 Contribution by Probability of Core Melt of
Accident Sequences Affected by Feed and Bleed*

Anal. No.	Sequence	Internal Event Core Melt Prob. w/o FNB	Contribution to Total Internal Event Core Melt Prob. (%)	Internal Event Core Melt Prob. w/FNB	Contribution to Total Internal Event Core Melt Prob. (%) of Replaced Sequences
69	S ₂ ML	2.19E-7	-	5.84E-10	-
70	S ₃ ML	8.34E-6	0.7	3.54E-6	-
71	T ₁ MPQL	3.72E-6	0.3	8.11E-7	0.9
72	T ₁ MPQL	1.46E-5	1.2	5.19E-8	0.1
73	T ₃ MPQL	6.20E-6	0.5	4.11E-8	-
74	T ₃ QL	7.22E-6	0.6	2.56E-8	-
75	T ₄ MPQL	4.36E-6	0.4	2.84E-7	0.3
76	T ₅ MPQL	2.83E-6	0.2	2.15E-7	0.2
77	T ₁ MPQL	9.74E-5	7.9	2.13E-5	24.3
78	T ₂ MPQL	5.04E-4	40.8	2.38E-6	2.7
79	T ₃ MPQL	2.40E-4	19.4	1.13E-6	1.3
80	T ₄ MPQL	1.92E-4	15.5	4.08E-6	4.7
81	T ₅ MPQL	1.04E-4	8.4	2.46E-6	2.8
		1.18E-3	95.1	3.28E-5	37.3
Total Core Melt Probability (internal events w/o FNB)					1.235E-3
Total Core Melt Probability (internal events with FNB)					8.758E-5
$\Delta p =$					1.147E-3

* See Tables B.15 through B.17 for details

Therefore, a feed and bleed capability reduces the estimated core melt probability for internal events at ANO-1 by $1.15\text{E-}3$ per reactor year. It is noted that this result is substantially greater than several other recent studies. For example, Sandia estimated a reduction in core melt probability of $4.0\text{E-}6$ when examining the cost-benefit of adding PORVs to System 80 plants.¹⁷ However, in that study common mode failures in pumps and valves were not modelled as extensively as they are in the present work.

In each case, with and without an FNB capability, the same five accident sequences dominant. Without FNB it is sequences 77-81, i.e., TMPQL. The corresponding sequences replaced are 15, 17, 19, 51, and 59, i.e., TMPLE. Without FNB these five sequences contribute 92.0% of the total core melt probability. With FNB the corresponding sequences contribute 35.8% of the total core melt probability. The causes in each case can be seen in the dominant cut sets for the ten accident sequences given in Tables 10.2-10.6. Recovery has been applied although the recovery actions are not shown.

The most obvious difference between the sequences with and without FNB is the core melt probabilities. The basic reason for this is the absence of FNB events in some of the cut sets. For example, in Table 10.2 DG2-GEN-LF does not need to fail to cause core melt without FNB. Also, the group of cut sets with EFWOP7AX-PTD-LF becomes doubles instead of triples. In Table 10.3 FNB-MANACT disappears without an FNB availability and also the EFWOP7AX-PTD-LF cut sets are doubles instead of triples. Table 10.4 for the T_3 accident sequence is similar to Table 10.3. Tables 10.5 and 10.6 also exhibit the FNB-MANACT difference as well as cut sets that were doubles given an FNB capability become singles without FNB. The end result is dramatic as the difference in $p(\text{CM})$ is $1.15\text{E-}3$.

An additional issue related to feed and bleed has been examined. As described elsewhere, the base case analysis for ANO-1 assumes feed and bleed procedures are available and would be used, and it further assumes that the ERV block valve is closed because that was the operating condition observed during plant visits. The influence of this assumption was examined by conducting an analysis with the block valves open. For the twenty FNB sequences affected by the block valve, the contribution to core melt probability is $1.8\text{E-}5$ with the valves closed and $1.8\text{E-}5$ with the valves open. ANO-1 is a B&W plant so failure of both SRVs would have to occur before the ERV would be required. This accounts for there being no difference between the blocked and unblocked ERV cases. Although recovery was not applied, there is no reason to believe that would change the conclusion. Thus, operating with the ERV blocked has no practical effect upon the probability of core melt for the initiating events covered in this analysis, but there could be an effect for ATWS sequences.

Table 10.2a Selected Cut Sets for Sequence 15 (Boolean 22)
 $T_1 MPQLE$ (FNB Available)

Core Melt Probability	Multiplier	Cut Sets
1.48E-5	$T_1 MP\bar{Q}$ *	BATT-CM
1.76E-6	$T_1 MP\bar{Q}$ *	DG-CM * EFWOP7AX-PTD-LF
1.70E-6	$T_1 MP\bar{Q}$ *	DG1-GEN-LF * DG2-GEN-LF * EFWOP7AX-PTD-LF
1.84E-6	$T_1 MP\bar{Q}$ *	EFWOP7AX-PTD-LF * 12 DOUBLES involving diesels, pumps, valves, and batteries

Table 10.2b Selected Cut Sets for Sequence 77 (Boolean 23)
 $T_1 MP\bar{Q}\bar{L}$ (FNB Not Available)

Core Melt Probability	Multiplier	Cut Sets
1.48E-5	$T_1 MP\bar{Q}$ *	BATT-CM
1.76E-6	$T_1 MP\bar{Q}$ *	DG-CM * EFWOP7AX-PTD-LF
4.47E-5	$T_1 MP\bar{Q}$ *	EFWOP7AX-PTD-LF * DG1-GEN-LF
2.64E-5	$T_1 MP\bar{Q}$ *	EFWOP7AX-PTD-LF * 14 SINGLES involving diesels, pumps, valves, and batteries
3.36E-6	$T_1 MP\bar{Q}$ *	EFWSP-CM
3.36E-6	$T_1 MP\bar{Q}$ *	EFW0019X-XOC-LF

Table 10.3a Cut Sets for Sequence 17 (Boolean 30)
 $T_2MP\bar{Q}LE$ (FNB Available)

Core Melt Probability	Multiplier	Cut Sets
8.22E-7	$T_2MP\bar{Q}$ *	EFWOP7AX-PTD-LF * FNB-MANACT * EFWOP7BA-PMD-LF
3.09E-7	$T_2MP\bar{Q}$ *	EFWOP7AX-PTD-LF * SWSP-CM
2.65E-7	$T_2MP\bar{Q}$ *	FNB-MANACT * EFWSP-CM
2.65E-7	$T_2MP\bar{Q}$ *	FNB-MANACT * EFW0019X-XOC-LF
1.55E-7	$T_2MP\bar{Q}$ *	LOSS-OSP * BATT-CM
5.64E-7	$T_2MP\bar{Q}$ *	EFWOP7AX-PTD-LF * 20 DOUBLES involving cooling, pumps, valves, CM, & FNB-MANACT

Table 10.3b Cut Sets for Sequence 78 (Boolean 31)
 $T_2MP\bar{Q}L$ (FNB Not Available)

Core Melt Probability	Multiplier	Cut Sets
2.69E-4	$T_2MP\bar{Q}$ *	EFWOP7AX-PTD-LF * EFWOP7BA-PMD-LF
8.82E-5	$T_2MP\bar{Q}$ *	ESWSP-CM
8.82E-5	$T_2MP\bar{Q}$ *	EFW0019X XOC-LF
3.66E-5	$T_2MP\bar{Q}$ *	EFWOP7AX-PTD-LF * 16 SINGLES involving cooling, pumps, valves, and buses

Table 10.4a Cut Sets for Sequence 19 (Boolean 30a)
 $T_3 \overline{MPQLE}$ (FNB Available)

Core Melt Probability	Multiplier	Cut Sets
3.84E-7	$T_3 \overline{MPQ}$ *	EFWOP7AX-PTD-LF & FNB-MANACT * EFWOP7BA-PMD-LF
1.47E-7	$T_3 \overline{MPQ}$ *	EFWOP7AX-PTD-LF * SWSP-CM
1.26E-7	$T_3 \overline{MPQ}$ *	FNB-MANACT * EFWSP-CM
1.26E-7	$T_3 \overline{MPQ}$ *	FNB-MANACT * EFWOO19X-XOC-LF
7.39E-8	$T_3 \overline{MPQ}$ *	LOSS-OSP * BATT-CM
2.68E-7	$T_3 \overline{MPQ}$ *	EFWOP7AX-PTD-LF * 20 DOUBLES involving cooling, pumps, valves, CM, and FNB-MANACT

Table 10.4b Cut Sets for Sequence 79 (Boolean 31a)
 $T_3 \overline{MPQL}$ (FNB Not Available)

Core Melt Probability	Multiplier	Cut Sets
1.28E-4	$T_3 \overline{MPQ}$ *	EFWDP7AX-PTD-LF * EFWOP7BA-PMD-LF
2.30E-5	$T_3 \overline{MPQ}$ *	EFWOP7AX-PTD-LF * 19 SINGLES involving pumps, cooling, valves, and CM
4.20E-5	$T_3 \overline{MPQ}$ *	EFWSP-CM
4.20E-5	$T_3 \overline{MPQ}$ *	EFWOO19X-XOC-LF
2.52E-6	$T_3 \overline{MPQ}$ *	TM-LOGIC-CM
3.78E-7	$T_3 \overline{MPQ}$ *	ACT-SG-SENS-CM

Table 10.5a Cut Sets for Sequence 51 (Boolean 38)
 $T_4 \overline{M} \overline{P} \overline{Q} \overline{L} E$ (FNB Available)

Core Melt Probability	Multiplier	Cut Sets
3.28E-6	$T_4 \overline{M} \overline{P} \overline{Q}$ *	EFWOP7AX-PTD-LF * 35 SINGLES involving cooling, pumps, valves, CM and buses
5.58E-7	$T_4 \overline{M} \overline{P} \overline{Q}$ *	FNB-MANACT * EFWOP7AX-PTD-LF
5.51E-8	$T_4 \overline{M} \overline{P} \overline{Q}$ *	EFWOP7AX-PTD-UTM * 3+ SINGLES involving pumps and valves

Table 10.5b Cut Sets for Sequence 80 (Boolean 39)
 $T_4 \overline{M} \overline{P} \overline{Q} \overline{L}$ (FNB Not Available)

Core Melt Probability	Multiplier	Cut Sets
1.86E-4	$T_4 \overline{M} \overline{P} \overline{Q}$ *	EFWOP7AX-PTD-LF
3.61E-6	$T_4 \overline{M} \overline{P} \overline{Q}$ *	EFWOP7AX-PTD-UTM
5.84E-7	$T_4 \overline{M} \overline{P} \overline{Q}$ *	EFW2802B-VOC-UTM
5.31E-7	$T_4 \overline{M} \overline{P} \overline{Q}$ *	EFW0019X-XOC-LF
5.31E-7	$T_4 \overline{M} \overline{P} \overline{Q}$ *	EFW010BX-CCC-LF
5.31E-7	$T_4 \overline{M} \overline{P} \overline{Q}$ *	EFWSP-CM
2.18E-7	$T_4 \overline{M} \overline{P} \overline{Q}$ *	EFWSV-CM
1.91E-7	$T_4 \overline{M} \overline{P} \overline{Q}$ *	EFW2802B-VOC-LF

Table 10.6a Cut Sets for Sequence 59 (Boolean 46)
 $T_5MP\bar{Q}LE$ (FNB Available)

Core Melt Probability	Multiplier	Cut Sets
1.60E-6	$T_5MP\bar{Q}$ *	EFWOP7AX-PTD-LF * 23 SINGLES involving cooling, pumps, valves, CM and buses
2.74E-7	$T_5MP\bar{Q}$ *	EFWOP7AX-PTD-LF * FNB-MANACT
7.83E-8	$T_5MP\bar{Q}$ *	D02-125VDC-LF
7.83E-8	$T_5MP\bar{Q}$ *	D21-125VDC-LF

Table 10.6b Cut Sets for Sequence 81 (Boolean 47)
 $T_5MP\bar{Q}L$ (FNB Not Available)

Core Melt Probability	Multiplier	Cut Sets
9.14E-5	$T_5MP\bar{Q}$ *	EFWOP7AX-PTD-LF
8.56E-6	$T_5MP\bar{Q}$ *	EFWOY2B-VCC-LF
1.77E-6	$T_5MP\bar{Q}$ *	EFWOP7AX-PTD-UTM
3.92E-7	$T_5MP\bar{Q}$ *	ACT-EFBB-LF
2.87E-7	$T_5MP\bar{Q}$ *	EFW2802B-VOC-UTM
2.61E-7	$T_5MP\bar{Q}$ *	EFWSP-CM
2.61E-7	$T_5MP\bar{Q}$ *	EFW0019X-XOC-LF
2.61E-7	$T_5MP\bar{Q}$ *	EFW010BX-CCC-LF

For seismic induced transients the following observations can be made. For the base case in which feed and bleed is available, the probability of core damage is $7.30E-5$ per reactor year. The dominant sequence is a transient with a loss of offsite power (~70%) involving failure of the EFWS and HPIS. The dominant contributors to this sequence are failures of different combinations of two electrical busses and either the CST or EFW motor-operated valve 2802. If modifications are made to upgrade the BWST, 4160 VAC and 480 VAC busses, the frequency of core damage drops to $8.83E-6$ per reactor year. Blackout type cutsets now dominate the transient sequence. In the base case the dominant failure modes involved combinations of electrical bus failures with or without another event to fail the EFWS. With seismic upgrades the failures are dominated by common mode failures of either pumps or valves and various combinations of support system failures involving the batteries, ventilation and service water. Thus, reduction of the seismic vulnerability enhances the feed and bleed capability.

In this investigation the addition of a dedicated feed and bleed system was not pursued for several reasons. First, if a dedicated system were added which was not completely independent (i.e., connections to RCS in all new piping with no reliance on existing valves, etc.), then it would be subject to failures such as operator errors and common mode failures of support systems. If a dedicated system without a separate dedicated power supply system were included, then it could still be subject to the failures in the emergency power system. This was illustrated in the results of the System 80 PORV Study.¹⁷ In that instance there was no benefit for response to LOSP transients. There was a benefit for SGTR and SBLOCA events because these events were dominated by local HPI failures which are countered by the added redundancy of a dedicated F&B system. Even so, the overall reduction in frequency of core melt in that study was only about a factor of 2.5. At ANO-1 the small break accident sequences to which F&B applies involve failures after a transient initiating event and so one would expect only limited, if any, improvement even with a dedicated system. (NOTE: In this TAP A-45 study tube rupture in a single steam generator is considered isolatable and therefore a transient initiating event. Simultaneous tube ruptures in both steam generators were not considered.) Second, the benefits of a completely independent feed and bleed system were not evaluated in the System 80 PORV Study.¹⁷ However, the costs associated with such a dedicated, independent system are substantial based upon what was done in the earlier study. A rough extrapolation of the available information suggests that an additional pump and associated hardware would cost in excess of \$6 million if the equipment were located in an existing structure. It was also estimated that a dedicated emergency electrical power system could run \$11 million. (This is consistent with current TAP A-45

estimates of costs associated with adding an emergency diesel generator at another site.) There would be substantial added construction costs if the dedicated pump were housed in a new separate structure. All of this would make the cost of a fully independent feed and bleed system 3 to 4 times that of other viable alternatives and a significant fraction of those for the add-on auxiliary feedwater/makeup system which removes decay heat without contamination of containment. Third, the ADHR study results¹⁸ indicate only about a factor of 2.5 improvement in core melt probability with an add-on HPI system that has a dedicated power supply. The add-on HPI was not as effective as an add-on AFW in reducing core melt probability at comparable cost. The add-on AFW system gave about a factor of 20 reduction in core melt probability. Given these earlier results and some of the timing and thermal hydraulic considerations outlined below, a dedicated F&B system was not examined for ANO-1. However, while this report was in progress, two concepts for a feed and bleed system were proposed by G. A. Reed of the Advisory Committee on Reactor Safeguards.¹⁹ These concepts will be examined in the overall context of TAP A-45 to evaluate their feasibility, costs and effect upon core melt probability and risk. Initial estimates are that $p(cm)$ might be reduced by factors of 10 to 20, depending upon the configuration adopted, but impact estimates are not yet available.

10.1.2 Impacts Associated with Feed and Bleed

In a situation such as that which exists at ANO-1 where feed and bleed can be used without installing new equipment, there are no direct capital costs. However, there are other impacts or implications of feed and bleed which need to be considered.

Thermal-hydraulic Phenomenology - Because of the broad interest in feed and bleed as a last resort decay heat removal method, questions have been raised regarding its actual viability in terms of the thermal-hydraulic phenomenology and potential environmental effects on equipment. Los Alamos National Laboratory (LANL) has examined the phenomenological questions very extensively under NRC sponsorship. The LANL results have been reported in a series of reports (References 13 through 15). The objectives of the LANL work were to:

- 1) Evaluate success or failure of feed and bleed in specific reactors and for specific accident sequences.
- 2) Identify plant specific and generic insights.
- 3) Use insights to predict feasibility of feed and bleed for PWRs for which detailed analyses were not performed.

In conducting their investigations LANL used audited models of four specific plants. In this instance audited implied that

all data is traceable to its source and that the input values are generally accepted as being realistic.

Specific response calculations were made by LANL for Calvert Cliffs 1, Zion 1, Oconee 1 and H. B. Robinson 2 using the TRAC-PF1 thermal-hydraulics code. It is also important to bear in mind the underlying assumptions and success criteria used by LANL. In References 18 and 19 LANL assumed that the plant equipment needed for F&B was available and fully operable throughout the transient, that is, equipment environmental qualification issues were not considered; nominal performance was used. In Reference 18 LANL further assumed that individual items of equipment were not functional, in one instance only 1 of 2 PORVs was used, in another only 1 of 2 HPI pumps. In those instances where the plant was being taken from reactor trip to hot standby, success was defined as attainment of stable RCS pressure above LPI and accumulator actuation levels, vessel mass inventory not decreasing and cladding temperature below or near saturation. When going from reactor trip to hot shutdown, success was defined as controlled RCS cooling and depressurization to LPI or RHR cut-in conditions.

A wide range of transient conditions was investigated by LANL including loss of feedwater, loss of offsite power, main steam line breaks, main feed line breaks, and steam generator tube rupture. These are described in detail in the referenced reports.

LANL concluded from this work that for the plants studied in detail, under the assumptions stated, feed and bleed would be successful if initiated no later than the time of steam generator dryout although the operating boundaries shift when less than nominal equipment is used. The actual time under nominal conditions varies from about 8 minutes for Oconee 1 (low inventory, once through B&W steam generator) to about 40 minutes (large U-tube Westinghouse steam generators).

Because ANO-1 is a B&W plant very similar to Oconee 1, it is appropriate to consider those results in more detail.

The following conclusions about feed and bleed at the Oconee 1 plant were reported in Reference 13.

- 1) A feed and bleed procedure can be used successfully to transition the plant from reactor trip to a hot holding condition, i.e., power generation has ceased but primary system pressures and temperatures remain at operating levels. If feed and bleed is initiated before or at the time of the generation of the containment overpressure signal, the primary system can be stabilized in a liquid-full and subcooled state. The transitions to hot holding can be successful even if only one of the two HPI pumps is used.

- 2) A feed and bleed procedure can be used successfully to transition the plant from reactor trip to a hot holding condition if initiated within 300 seconds of containment overpressure. However, some boiling will occur in the core. Feed and bleed may prevent core damage if initiated at times greater than 300 seconds, but significant voiding would occur.
- 3) A feed and bleed procedure can also be used to transition the plant to hot shutdown. However, this conclusion is based upon the assumption that a capability exists to replenish the inventory in the BWST or to operate the LPI system, taking suction from the containment sump with delivery to the HPI system inlet.
- 4) Use of feed and bleed in the transition to hot shutdown could be simplified with increased PORV-relief capacity. With sufficient capacity (at least double the existing PORV capacity) the plant can be depressurized and cooled using only the inventory of the BWST.

Comparing key parameters (see Table 10.7) for Oconee 1 and ANO-1, it is reasonable to conclude that feed and bleed can be conducted successfully from the phenomenological point of view.

Table 10.7 Selected B&W Plant Parameters

<u>Parameters</u>	<u>Oconee 1</u>	<u>ANO-1</u>
Power (MW_t)	2568	2568
PORV Capacity (lbs/hr/ MW_t) (lbs/hr)	41.7	39.9
Number of PORVs	1	1
HPI Shut-Off Head (psi)	3034	3034
HPI Capacity gpm @ 1000 psi gpm @ 1600 psi	500 450	500 450
SG Dryout Time (full power min)	0.48	0.48

In summary, these analyses indicate that feed and bleed can be successful in plants provided the process is initiated early enough. The initiation time to prevent core uncover can be as little as 3 to 5 minutes into an event depending upon when reactor scram occurs. It would appear that this puts a considerable pressure upon the operators. A decision to open the primary must be made before there is really any time to

establish reasons for other equipment not functioning. It is noted that one of the concepts proposed by Reed¹⁹ alleviates this concern by blowing down to, and recirculating from, a closed system.

10.1.3 Operational Issues

Another aspect of the application of feed and bleed must also be considered. There has always been some question about the willingness of operators to open the primary system at early times given the history of education which says maintain primary system integrity. This concern seems to be borne out based upon recent reports on the June 9, 1985, incident at Davis-Besse, in which operators may have delayed initiating feed and bleed.²⁰

10.1.4 Environmental Qualification Concerns

In addition to the questions related to the thermal-hydraulic phenomenology, there are concerns about equipment qualification, that is, will the necessary equipment function under the environments that may be created.

Using the LANL calculations to define the mass and energy released into containment pressure/temperature profiles were developed by the NRC staff as a function of containment system performance. A large dry containment was used for the calculations. The general findings of that study as quoted in Reference 21 are:

"Following a feed and bleed transient if all the active containment heat removal systems fail, the containment pressure and temperature could exceed the design values, but would level off before the containment pressure reaches the ultimate containment strength of more than two times of design pressure. The containment pressure and temperature conditions following a feed and bleed transient are lower at the peak and at some later time (10^4 sec) than those following a LOCA. The conditions from feed and bleed could be slightly higher than those from LOCA during a period of several thousand seconds and at very later time (10^4 sec) when the elevated pressure and temperature are not significantly higher than those during normal operating conditions. The implications of this period to equipment qualification, during which the containment pressure and temperature conditions of feed and bleed are slightly higher than those from LOCA, is plant dependent and instrument specific."

At this point in time the qualification profiles for specific pieces of equipment have not been examined, and there are some unanswered questions related to the differing pressure-time and temperature-time profiles. Those of the LOCA environments peak rapidly and then drop off,²² while the F&B environments, although lower in peak amplitude, exist for longer periods of

time. The effect upon equipment performance has not been evaluated.

10.1.5 Summary

The preceeding analysis clearly shows that a feed and bleed capability can have a significant effect upon the probability of core melt. Also, the thermal-hydraulic studies certainly suggest that feed and bleed is possible from a phenomenological point of view. However, these apparent benefits must be viewed in the light of: a) the very stringent time constraints revealed, as decisions to feed and bleed must be made early in the accident progression for it to be successful; and b) the apparent operator reluctance to "open" the primary system. These latter concerns may be alleviated by a closed system such as has been proposed by Reed.

10.2 Secondary Side Blowdown

In this ANO-1 analysis, the term secondary blowdown, is used to describe the use of the atmospheric dump valves to rapidly depressurize the secondary side of the steam generators. This action is accompanied by primary side depressurization due to the increased primary to secondary heat transfer and subsequent cooling of the primary. The goal is to reach low pressure injection set points or residual heat removal cut-in conditions so that the system can be put on closed cycle cooling.

There has been some discussion related to the use of pressurizer sprays in conjunction with the rapid secondary depressurization to depressurize the primary after a loss of offsite power transient. In addition to an examination of the available literature on secondary side blowdown, Los Alamos National Laboratory (LANL) was asked to make some specific calculations. The LANL results are discussed here along with some insights from the probabilistic assessment.

10.2.1 Value of Secondary Side Blowdown

In the internal event analysis of the ANO-1 Nuclear Plant, the event trees were constructed to include the use of secondary side blowdown (see Section 2 and Appendix B). The success criteria included the use of both the atmospheric dump valves and the pressurizer sprays in this process. As shown in Appendix B, the use of secondary side blowdown influences 22 potential accident sequences. If a secondary side blowdown capability does not exist, then these 22 sequences reduce to 8 sequences. The specifics of the sequences replaced are discussed in Appendix B, but some comments are included here. The combined core melt probabilities for the affected sequences, prior to applying recovery actions to these sequences are:

P(cm) (affected sequences without blowdown) 1.88E-4
P(cm) (affected sequences with blowdown) 9.77E-5

On this basis the secondary side blow-down capability reduces the estimated core melt probability at ANO-1 by 9.0E-5. This is a smaller effect than that attributed to the feed and bleed capability (see Section 10.1.1 above). If secondary side blow-down and pressurizer sprays are not available the probability of core melt p(cm) (internal events only, but with feed and bleed), would be about 1.8E-4. If neither feed and bleed or secondary depressurization is available p(cm) rises to about 1.3E-3.

In the case of feed and bleed the overall gain (1.1E-3) was significant because the required systems with or without feed and bleed had different vulnerabilities. The improvement with secondary side blow-down is less dramatic (~ 9E-5) because differences in the required support systems with and without blow-down are not as pronounced. This can be illustrated by considering the SBLOCA sequence $S_3\bar{M}\bar{L}E$ which contributes about 87% of the core melt probability without blow-down cited above. If secondary side blow-down is feasible, there are three possible sequences; $S_3\bar{M}\bar{L}EH_1$, $S_3\bar{M}\bar{L}ED_2$ and $S_3\bar{M}\bar{L}XE$. In all of these sequences the initiation of the SBLOCA (S_3) is accompanied by loss of main feedwater (M) but auxiliary feedwater is available ($\bar{L}$). In the first instance the probability of core melt is influenced by the failure probability of high pressure injection (E). In the latter three sequences the combined probability of failure of: high pressure injection (E) and low pressure recirculation (H_2'); high pressure and low pressure injection (ED_2); and secondary side blow-down (X) and high pressure injection (E), respectively are the issues of concern. The sequence $S_3\bar{M}\bar{L}EH_1$ is a negligible contributor, while $S_3\bar{M}\bar{L}ED_2$ and $S_3\bar{M}\bar{L}XE$ contribute 91% and 8%, respectively to the combined probability. Therefore, the principal comparison is between sequences $S_3\bar{M}\bar{L}ED_2$ and $S_3\bar{M}\bar{L}E$. In sequence $S_3\bar{M}\bar{L}ED_2$, given the initial conditions $S_3\bar{M}\bar{L}$, two injection systems must fail to cause core melt, while in sequence $S_3\bar{M}\bar{L}E$ failure of only the high pressure system is enough to cause core melt. While these two situations have many cut sets in common, there are additional failures involving pumps and valves which can cause failure of the low pressure injection. Therefore, having a secondary blow-down capability does improve the core melt probability by about 51% whereas the feed and bleed improvement was about 92%.

10.2.2 Systems Required for Successful Depressurization

Several years ago EG&G (INEL) investigated small break accident mitigation at Zion 1, assuming coincident failure of both charging and high pressure injection.²³ The conclusions of that study were:

"The largest break diameter which will prevent a depressurization to the accumulator injection pressure is about 6 cm (2.4 cm).

"The small break sequence with coincident failure of charging and HPI rapidly leads to a core uncovering but sequence timing is directly affected by break diameter.

"The opening of a single atmospheric dump valve is not effective in preventing core uncovering.

"The opening of all four ADVs at 10 minutes has been shown to be an effective means of preventing core uncovering following 2.54 cm (1 in) and 5.08 cm (2 in) diameter breaks."

In these INEL investigations for Zion 1 no pressurizer sprays were used. The atmospheric dump valves had a flow area of 0.01015 m^2 (0.1092 ft^2) each, sufficient to pass 112.1 kg/S (247 lbs/sec) of saturated steam at an upstream pressure of 7.72 MPa (1120 psia).

Similar calculations were performed by Combustion-Engineering (CE) to address the question of how CE System 80 plants handle small break LOCAs without PORVs.²⁴ The calculations were performed for a 3410 Mwt class plant. Aggressive secondary side cool down at 100°F/hr was initiated 15 minutes into the event. Results indicated that the Safety Injection Tanks (SIT) begin to inject at about 3500 seconds. SIT injection continues until the RCS pressure reaches 100 psi. The low pressure injection pumps (LPI) will begin to inject at 200 psi. Based upon inspection of the graphical results in Reference 22, LPI operation should begin between 6000-7000 seconds. The ADVs used had a capacity of 195.2 lbs/sec at 900 psia with an effective flow area per valve of 0.108 ft^2 . There was one ADV per steam generator.

J. D. Harris of the Nuclear Operations Analysis Center at the Oak Ridge National Laboratory examined atmospheric steam dump systems for a number of PWRs.²⁵ The specific initiating events of concern are not spelled out in the Harris study, but the text implies that LOCAs and transients are considered. Harris does specifically state that he is concerned about heat removal after Main Steam Isolation Valve closure after some Design Basis Event. Based upon his relatively simple analysis of 11 PWRs, Harris concluded that: 1) two plants (Indian Point 2 and San Onofre 2 & 3) could achieve RHR entry using available auxiliary feedwater; 2) three plants (Fort Calhoun, Calvert Cliffs 1 & 2, and Davis Besse 1) could not and; 3) six (Palisades, Salem 1 & 2, Crystal River 3, Beaver Valley 1, Millstone 2, and McGuire 1) could not if there were steam dump valve failures. Thus, the Harris approach agrees with the CE analysis that the 3410 Mwt class can depressurize. Although the previous estimates by both INEL and CE suggest that

existing ADVs are adequate to reach RHR conditions, the questions raised by Harris prompted further examination.

Los Alamos National Laboratory (LANL) was asked to examine the question using the TRAC/PF1 thermal-hydraulics code. The Calvert Cliffs plant model was used for this examination for several reasons. One, Calvert Cliffs was one of the plants Harris indicated could not reach RHR cut in. Two, this plant model has been used in several prior studies. Three, Calvert Cliffs was the reference plant for the cold shutdown investigations in TAP A-45.²⁶ LANL examined the use of ADVs in response to a small break LOCA with failure of high pressure injection²⁷ and to a loss-of-offsite power transient.²⁸ The major conclusions from the LANL work are as follows. From Reference 27 for the small break LOCAs:

"The primary system will cool and depressurize to the low-pressure injection system operating pressure and design temperature limits using only water supplies from the safety-grade condensate water storage tank.

"The procedure is successful even with additional equipment failures such as the failure of one atmospheric dump valve to open or the safety injection tanks to discharge. Such failures lengthen the time to start the low-pressure injection system.

"Because the depressurization rate is enhanced by the small break, the cooldown rate defines the time to low-pressure injection system operation.

"Doubling the auxiliary feedwater flow increased the cooldown rate but the increase was small to moderate.

"No calculation was extended to a low-pressure injection system operating temperature of 422 K (300°F). The cooldown from the design temperature limit to the operating temperature was very slow. Because the secondary-side pressure had reached a quasi-steady state it seems that further cooldown would track the decrease in decay power."

The LANL results for SBLOCA at Calvert Cliffs are consistent with the INEL Zion 1 study (Reference 1). The depressurization of the primary is "assisted" by flow out of the break and there is sufficient inventory available to adequately cool down. It would appear that a similar situation should exist at ANO-1 because there the two ADVs have a combined capacity of $1.33\text{E}6$ lbs/hr as compared to $1.74\text{E}6$ lbs/hr for Zion. But, detailed calculations have not been performed for ANO-1.

Similarly, from Reference 28 for the LOSP transient:

"Cooldown and depressurization to RHR entry conditions does not occur for the case where operator uses only existing plant ADVs and AFW.

"If the operator uses APS in addition to existing ADVs, depressurization is improved, but RHR entry is still unattainable, because of secondary side cooldown and depressurization limitations.

"Doubling the ADV capacity still may not provide sufficient flow capability to bring secondary pressure and temperature down low enough to allow primary cooldown to RHR entry conditions.

"By increasing the ADV capacity to 400% [of nominal] the plant can reach RHR entry conditions within 3-1/2 hours from the initiating event. A 500% flow reduces this entry time to about 3 hours."

With regard to the LOSP transient, after it was ascertained by LANL that the APS was required for depressurization using only one ADV per steam generator, the APS was used in all subsequent calculations. Therefore, the ability to depressurize Calvert Cliffs with larger relief capacity but without APS was not addressed. However, to put some perspective on this question, it is noted that the ANO-1 LOSP sequence that could be affected by the secondary blow-down capability does not contribute to the core melt probability discussed in Section 10.2.1 above. Thus the response to the LOSP transient is not a major issue. It is our understanding of ANO-1 staff that they could reach cold-shutdown with existing CST inventory given a LOSP.

10.2.3 Summary

In summary, it appears reasonable to state that: 1) the secondary side blowdown capability only has a demonstrable effect upon the probability of core melt given SBLOCA events with failure of HPI and aggressive cooldown using the ADVs or steam dump valves and pressurizer sprays reducing the primary pressure to RHR entry conditions. Successful cooldown to RHR entry conditions using only ADVs after a LOSP transient has not been fully explored, but in the case of ANO-1 this sequence is not a major contributor to the core melt probability.

11.0 SUMMARY, CONCLUSIONS, AND OBSERVATIONS

This report documents the investigation of the adequacy of decay heat removal (DRH) at the ANO-1 Nuclear Plant. This particular plant was selected as an example plant for several reasons. One, the results of an initial qualitative screening process suggested that there were some potential DHR vulnerabilities that deserved further study. Two, it is "typical" of the layout and construction practices of a number of plants of similar vintage. Three, the utility management, Arkansas Power and Light Co., expressed a willingness to participate in this aspect of the USI A-45 investigations. The following sections highlight some of the findings and insights including non-quantifiables, discuss aspects of DHR issues not addressed specifically, and finally make some generic observations.

11.1 The DHR Assessments

This program was not designed to conduct full-scope probabilistic risk assessments (PRA) of the plants being studied. However, it was established very early in the program that limited-scope assessments would be necessary to adequately address the issues. Because TAP A-45 is considering the full range of potential failure mechanisms, the study includes a PRA based examination of internal random and systematic faults and a simplified PRA based examination of special emergencies. This study uses the term "special emergencies" to describe those events such as fires, floods, winds, etc., which are often labeled external events elsewhere. These latter investigations are characterized as "simplified" because many of the analysis techniques are still under development and therefore, in some instances, have substantial subjective engineering judgment included.

11.1.1 The Internal Analysis

The details of the internal analysis are presented in Section 2 and Appendices A and B. From this analysis it was established that the random internal failures (including recovery actions) had an associated core melt probability of $8.8\text{E-}5$. There were eight internal vulnerabilities identified, the effects of which accounted for 82% ($7.2\text{E-}5$) of core melt probability. Equipment and system modifications were proposed for these eight vulnerabilities which reduce the corresponding core melt probability to $2.6\text{E-}5$.

11.1.2 The Special Emergency Analysis

The details of the special emergency analyses are presented in Section 3 and Appendices C through I. The special emergencies included in this study are: seismic events, fire, internal and external floods, extreme winds, lightning and internal sabotage. The seismic contribution to core melt was estimated to be

7.3E-5. Contributed primarily by two specific vulnerabilities. The fire contribution to the core melt probability was 5.8E-6 attributable to one specific vulnerability. Extreme wind contributed 5.3E-2, all of which was attributed to a single vulnerability. Equipment, system and/or building modifications were proposed to eliminate or reduce these vulnerabilities. Internal flood, external flood, and lightning contributed $< 1.0E-7$, $7.2E-6$ and $1.8E-7$, respectively, to the core melt probability. No modifications were proposed to counter specific vulnerabilities from these three initiating events. The sabotage analysis was not quantified, but modifications identified for several special emergencies will be helpful in countering potential sabotage.

11.1.3 Assumptions and Bases for the Analyses

It seemed appropriate to assemble in one place for reference the assumptions or other special factors upon which this case study is based. This is done here under six headings: internal analysis, external analysis, dedicated add-on system, impact analysis, consequence analysis, and value/impact analysis. While these lists may not be complete, they should describe most of the important bases for the analyses that may affect the results.

11.1.3.1 Internal Analysis Assumptions/Conditions

GENERIC

1. System configurations and equipment locations were derived from the FSAR, drawings supplied by the utility, plant visits, and communications with plant personnel.
2. Operational procedures in general were not analyzed in detail and we relied upon statements from plant personnel regarding procedures.
3. The scope of initiating events was limited to small LOCA and transients including loss of an AC bus or a DC bus but excluding the loss of service water initiator. ATWS was not considered.
4. System success criteria used are generally conservative.
5. Containment failure and sump flashing do not result in ECCS failure due to pump cavitation.
6. Given success of the EFWS, RCS makeup is not required for transients.
7. Transient induced LOCAs cannot proceed to core melt due to failure of the safety relief valves to open since by definition the valve is already stuck open.

8. Even if the pressurizer safety relief valves are not required to provide RCS overpressure protection, these valves have a 7% chance of opening anyway.
9. The frequency of loss of offsite power quoted in NUREG/CR-2728, i.e., $8.6E-2$ per reactor year, was used rather than other suggested generic or plant specific numbers. This provides a consistency between analyses.
10. Various component failure probabilities were derived in these analyses and may differ from other data bases.
11. Circuit breaker failures were grouped with the associated buses for simplicity.
12. Common mode pump and valve failure probabilities were derived based upon ASEP information.
13. Control circuit failures were combined with the local fault probabilities for pumps and motor operated valves.
14. Recovery of the PCS was only permitted for selected accident sequences depending on the availability of power and whether the initial loss occurred as part of the initiating event or subsequent to it.
15. Only one recovery action was allowed in addition to recovery of LOSP or PCS, except when the allowable recovery time is greater than 2 hours, 2 recovery actions were permitted.
16. Cut sets with two UTM events from the same system were considered invalid.
17. Hardware failures of pumps and valves were not considered recoverable except for that part of the failure attributed to control circuit failures for valves which did have a recovery applied.
18. Normally-closed motor-operated valves in pump suction lines were considered recoverable since it was assumed operators would shut down the corresponding pumps before cavitation occurs based on discharge pressure and flow indication, thus making the recovery meaningful.
19. Actuation system failures are assumed to be remotely recoverable by manual initiation of the particular system.
20. Successful safety injection extends any accident sequence time to core melt by approximately 3 hours.
21. Recovery was only applied to the top cut sets of the top accident sequences--approximately 95%.

SPECIFIC

1. Emergency Feedwater System (EFWS) success is sufficient for core cooling.
2. Feed and Bleed is a viable mode of operation and no containment overpressure protection is required in that mode.
3. The HPIS with backup from the SWS provides cooling for the RCS pump seals, however, this was not modeled and it was assumed the seals do not require cooling.
4. Recirculation is not required for transient initiating events due to the size of the BWST.
5. The HPIS must be manually started for transients and S₃ LOCA.
6. Emergency Core Recirculation (ECR) cooling is required for the continued operation of the high pressure pumps.
7. Secondary blowdown mode of operation requires pressurizer sprays and either the steam bypass valves or both atmospheric dump valves.
8. Containment overpressure protection for LOCAs requires either 1 of 4 fans or 1 of 2 spray injection and recirculation with high pressure recirculation and sump cooling. While certain sequences could require less, this criteria was used in all cases.
9. The Electromatic Relief Valve (ERV) was assumed to be blocked.
10. Long term station blackout was not a problem for ANO-1 due to 8 hour batteries and a large CST.
11. The diesel generators do not require DC for field flashing but do require DC for instrumentation and remote control or startup.
12. There were two diversion path failures modeled for the SWS that could fail the system functions.
13. The third SWS pump was not modeled but a recovery action was applied to applicable cut sets to account for its use.
14. Switchgear and battery room cooling failures can be recovered by opening doors and possibly setting up portable fans to circulate the air.

11.1.3.2 External Analysis

General

A number of external events (special emergencies) were excluded because they were beyond the scope and depth of our analysis to examine carefully or because of one of the following reasons.

- a. The event is of equal or lesser damage potential than the event for which the plant has been designed.
- b. The event has a significantly lower mean frequency of occurrence than other events with similar uncertainties.
- c. The event cannot occur close enough to the plant to affect it.
- d. The event is included in the definition of another event.
- e. The event does not directly affect the decay heat removal system.
- f. The event is slow in developing and there is sufficient time to go to cold shutdown.

This left seven events for further consideration for which the corresponding assumptions are given below.

Seismic

1. Seismic hazard curves were taken from other studies and not developed specifically for this program.
2. Since drawings were not available for the BWST it was assumed that it was designed to the procedures in TID-7024 which can lead to a non-conservative design in some cases. Thus, using past experience, a median fragility level of 0.40 g was assigned to the BWST.
3. It was assumed that loss of DC fails the turbine driven emergency feedwater pump.

Fire

1. Only fires in a single room were considered.
2. Fire frequency and suppression reliability was based on available industry data.
3. Two transient combustible exposure fires were assumed to bound all transient and electrically initiated fires.
4. It is assumed that a fire occurs independently of any other initiating event and the operators will always scram the reactor once they know there is a fire.
5. Passive components will be assumed to be unaffected by a fire.
6. Offsite power is assumed to be available during the fire unless the fire is capable of causing a LOSP.
7. Fire growth, spread, and suppression were treated independently.
8. No core melt timing was considered in the analysis.
9. When exact locations of cables could not be accurately determined, their locations were assumed.
10. Credit was given in the calculations for radiant energy shields, conduit, solid bottom cable traps, and completely enclosed cable trays.

Internal Flood

1. Internal flood frequencies, component damage thresholds, and physical room considerations were based on previous industry wide data bases.
2. It is assumed that an internal flood occurs independently of any other initiating event and the operators will always scram the reactor once they know flooding is occurring.
3. Flood barriers always maintain their structural integrity.
4. Flooding via drains backing up into other areas was not considered.
5. Only single flood sources were analyzed.
6. Components fail from submergence in water to a critical level or from direct spray from the flood source. It is assumed that switchgear and buses will fail if they are

standing in six inches or more of water. An electric motor is assumed to fail when the water level reaches the bottom of the motor casing.

7. Offsite power is assumed to be available during internal flood when the flood is capable of causing an LOSP.
8. No core melt timing was considered in the analysis.

External Flood

1. It is assumed that loss of offsite power will occur during an external flood event.
2. Flooding due to rain or snow melt would be slow developing and thus not present any threat.
3. Equipment failure is assumed given submergence.

Extreme Wind

1. Equipment protected by concrete structures at least 12 inches thick are adequately protected against tornado missiles.
2. All openings were either protected by barrier walls, tornado missile barriers, or if less than 100 square feet were considered too small to be vulnerable due to a low mean frequency of impact.
3. Loss of offsite power is assumed in extreme winds that could damage the plant.
4. Recovery of the diesel generator is possible, given bent exhaust stacks, by cutting the stacks to allow air intake.

Lightning

1. Lightning protection equipment was not specifically evaluated as to proper installation or maintenance.
2. It is assumed offsite power fails with a likelihood of 0.5 when lightning strikes the plant.
3. A generic ground flash density for the southern U.S. was used in the analysis.
4. It was assumed from the literature review, site inspection, and discussions with plant personnel that the plant was designed according to NRC recommendations to be able to withstand a lightning strike of 200 kA.

5. The ANO site was estimated to be 1/4 square kilometer in area.

Sabotage

This is discussed in Appendix I under separate cover.

11.1.3.3 Dedicated Add-on Analysis

1. The system selected was based on the results of previous NRC studies, and the assumption that forced or natural circulation through the steam generators provides the most effective path for decay heat removal from the core.
2. Offsite power will be used if available but a completely independent power system is included.
3. All structures and connections are designed to seismic Category I specifications. All equipment is assumed to be designed to withstand all special emergencies.
4. Common mode between add-on trains was assumed negligible.
5. Pipe rupture in the add-on system was considered insignificant.

11.1.3.4 Impact Analysis

1. All costs are in terms of 1985 dollars.
2. Replacement power, if required, in order to implement an alternative is in 1985 dollars.
3. In general, specific location for equipment and pipe runs were considered. In a few cases the space available may be uncertain.
4. Local labor rates were considered for all tasks.

11.1.3.5 Consequence Analysis

1. The mapping of accident sequences and containment system failures to release categories was accomplished using results of past studies adapted to TAP A-45 so no specific phenomenological computer runs were accomplished.
2. Containment failure mode probabilities were updated in light of recent studies to allow for a significant probability of containment success.
3. A range of offsite population doses were considered since the source term issue has not been settled.

4. No full scale uncertainty analysis was performed. Various issues are discussed in Section 8 considering the possible range of values for the results.

11.1.3.6 Value-Impact Analysis

1. The value/impact analysis relied heavily upon NRC documents suggesting various performance measures. This includes the concepts of negative values and negative impacts.
2. A 5% discount rate was used.
3. A \$1000/person rem value was used.
4. The plant lifetime was based upon the remaining period left in the license from the time of the study.

11.1.4 Base Case Results

When the results of the analyses are combined, the base case core melt probability is $1.79\text{E-}4$ made up of the following contributors:

Internal	$8.8\text{E-}5$
Seismic	$7.3\text{E-}5$
Fire	$5.8\text{E-}6$
External Flood	$7.2\text{E-}6$
Extreme Wind	$5.3\text{E-}6$
Lightning	$1.8\text{E-}7$
	$1.79\text{E-}4$

11.1.5 The Alternatives and Revised Core Melt Probabilities

As noted above, specific modifications were recommended for the most significant vulnerabilities identified. These modifications were combined to define one alternative which could be implemented at the plant (see Section 5.0). The impacts, costs, associated with implementing the alternative were estimated using standard industry practices (see Section 6.0). The core melt probability was then reevaluated assuming the alternative was in place (see Section 7.0) with the following results:

<u>Alternative</u>	<u>Probability of Core Melt per Reactor Year</u>	<u>Change from Base Case</u>
1	$5.14\text{E-}5$	$1.28\text{E-}4$
2	$1.45\text{E-}5$	$1.65\text{E-}4$

11.1.6 The Value-Impact Summary

The results of the core melt probability estimates were combined with the results of an analysis of containment systems performance and CRAC-2 calculations of offsite consequences to establish estimates of public risk (see Section 7.0). These estimates and the impact results were then combined in a value-impact analysis. Detailed results are tabulated and presented in Section 9.0. Table 11.1, which is the same as Table 9.6, summarizes the key results.

11.1.7 Discussion of the results

There are a number of conclusions or observations based upon Table 11.1 which should be highlighted.

- a) There appears to be a number of feasible modifications reflecting both internal and special emergency concerns which, if implemented, can significantly reduce the estimates of core melt probability. This tends to confirm one of the original assumptions of the program, i.e., there are some DHR vulnerabilities in the existing plants and they can be countered.
- b) The total averted population dose is very modest in all cases, on the order of a few hundred person rem. However, the averted dose ratio indicates that the alternatives can be effective in reducing population dose. For example, alternative 2 eliminates over 90% of the exposure predicted in the base case.
- c) Considering only offsite costs the value impact ratio and net benefit measures both suggest that the alternatives lack cost effectiveness. Also, the dollars per person-rem measure exceeds the nominal \$1000 per person-rem.
- d) If the value measures include the onsite costs that can be averted (personnel exposure, replacement power, cleanup, and loss of investment) the value measures still suggest a lack of cost effectiveness.
- e) In terms of the quantitative measures alternative 2, which is the add-on independent train, does not fare well because it involves significant (tens of millions of dollars) of capital expense. However, it does provide other non-quantifiable attributes (see Section 11.2) which must be considered.

11.2 The Add-on Decay Heat Removal System

11.2.1 Non-quantifiable Value Issues

As noted above, the add-on system capital expense puts the system at a disadvantage in a simple quantitative analysis.

TABLE 11.1 ANO-1 - SUMMARY OF VALUE-IMPACT MEASURES (Central Value)

Alternative No.	Change in Core Melt Probability	Offsite Averted Dose (p-rem)	Total Averted Dose Ratio	V-I Analysis of Offsite Costs			V-I Analysis Based on Offsite and Onsite Costs		
				V-I Ratio	Net Benefit (\$ x10 ⁻⁶)	Dollars per p-rem	V-I Ratio	Net Benefit (\$ x10 ⁻⁶)	Dollars per p-rem
	Δp_m	V_2	ADR_n	VIR_0	NBV_0	DPR_0	VIR_n	NBV_n	DPR_n
1	1.28E-4	167	0.81	6.51E-3	-15.25	9.2E+4	1.62E-2	-11.24	3.7E+4
2	1.65E-4	218	0.95	2.21E-3	-59.23	2.7E+5	4.07E-3	-54.07	1.4E+5

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However, there are other attributes which must also be considered. The add-on system, because of its complete independence from other plant systems, will have a significant effect upon residual risk. It is unaffected by fires and/or floods which might occur in other plant buildings. Similar comments can be made about this system with respect to equipment qualification issues. The add-on is protected from steam line breaks and high to medium pressure line breaks which could occur in the plant. Furthermore, since it provides the independent backup to other systems, a variety of potential failures of normal safety systems are covered. Because it is housed in a separate structure and interfaces only with the primary system inside containment, the add-on provides additional protection against sabotage. Since all the equipment is in standby status and is a last resort backup to existing safety systems, added security (access control) is possible without a major impact upon routine day-to-day operations.

11.2.2 Comparison with European Practice

For simplicity in design and costing the details of the add-on system proposed here were extracted directly from prior work.¹¹ Therefore, it is a single train system. In the earlier work this approach was adopted based upon the philosophy that it is a "last resort" system, and therefore a number of redundant and diverse systems would have to be failed before it is used. In contrast, in a number of European installations of a similar nature redundant trains are used.¹² However, their approach must be viewed within the general context of nuclear safety in Europe. In their "front line" safety systems they usually adopt the N+2 philosophy as compared to the N+1 approach in the United States. Therefore, they generally provide more redundancy. It should also be noted that in many instances their special emergency concerns are driven by the higher population densities near the sites, and the higher frequency of some events, e.g., aircraft crashes.

There is no doubt that increasing the redundancy would increase costs. Because time and resources originally led this program to the use of an existing design, cost data for a multiple train system is not available. These aspects will be considered in subsequent plant analyses for TAP A-45. Costs are also affected by the level of hardness selected for the structure housing the add-on system. The design here is for a Seismic 1 structure with limited access. Therefore, the add-on system will be available as long as the primary system it is designed to protect is available. This is generally consistent with the approach in Europe although there are instances there, notably in Switzerland, where the add-on is housed in a bunker with much greater hardness than the rest of the plant. Such bunkering may offer some advantage in coping with an external sabotage threat (which is not being considered in TAP A-45).

but does not appear significantly more beneficial in coping with insiders than the system proposed here.

The design used here does not include provisions for putting the plant into the residual heat removal mode from the add-on system. Operators would have to take control of the usual plant systems either locally or from the control room. A number of the European plants do have provisions within the add-on system building for taking control of the normal plant systems and proceeding to cold shutdown. In several instances this was simply based upon the philosophy that if you are having problems, the safest condition to be in is cold shutdown. Subsequent program investigations will examine the costs and design ramifications associated with adding such a capability.

11.3 Observations from the Case Study

Although the objective of TAP A-45 is to resolve decay heat removal issues in a generic way, it is difficult to reach broadly applicable conclusions based upon one plant. However, there are a number of observations which can be made.

- a) This study was consistent with other probabilistic risk assessments in that failures or vulnerabilities in support systems are important mechanisms, however, direct failures of frontline systems are dominant. See Table B.24 in Appendix B for further details.
- b) Emergency electric power availability again appears to be an important issue. This again emphasizes the close ties between USIs, e.g., A-44 and A-45. Other plants with minimal AC and DC systems are likely to benefit from modifications similar to those proposed for ANO-1.
- c) Common mode failures of pumps and valves have a significant contribution to the core melt probability.
- d) Most of the vulnerabilities identified can be reduced by modifications which involve little or no radiation exposure to construction personnel. This reflects the fact that the important vulnerabilities are in systems which are predominantly outside containment. Of course, the independent add-on requires access to the primary system and thus installation incurs some exposure.
- e) By careful planning and scheduling the sort of alternatives proposed here can be implemented during normal refueling outages, although the add-on might have to be spread across several outages.
- f) This case study suggests that the ability to feed and bleed on the primary system which is often considered a last resort

approach for removing decay heat, has very significant effect upon core melt probability or risk. However, thermal-hydraulic studies conducted in support of TAP A-45 raise serious questions as to the "last ditch" nature of feed and bleed. In order for it to be effective in preventing core uncover and melt, a decision to use it must be made quite early in the accident sequence.

A more general observation, which may be premature until several plant studies are complete, is that the methodologies of PRA and systematic analysis of special emergencies can be combined effectively in a relatively short time scale to gain significant insights into nuclear plant DHR reliability. It is believed that the techniques developed here to do a simplified seismic analysis will have application beyond TAP A-45.

11.4 Core Melts at Multi-Reactor Sites

11.4.1 Introduction

Several TAP A-45 plants, including ANO-1, share their sites with a second reactor. These multi-reactor sites often share support systems such as electric power and auxiliary feedwater. We have found that many of the dominant cut sets computed for one unit will actually affect both units and in some cases simultaneous melts could occur. For example, cut sets involving an extended loss of offsite power could very well lead to core damage at both units. Additionally, modifications designed to benefit a single unit could have similar positive effects on the second unit.

The effects of accidents which could cause both units to melt have not been addressed explicitly in this study. In the baseline analyses, a single reactor has been analyzed and shared systems only modeled where necessary. Multi-reactor effects at ANO-1 have, however, been examined as a sensitivity issue. In our approach, estimates of the total probability of a single or dual unit core melt at a multi-reactor site, costs and effects of the proposed DHR modifications on both reactors and the consequences from accidents at a site have been used to estimate the value-impact of such site modifications.

From our observations, most two-unit melt scenarios occur after initiating events which more or less uniformly affect both units such as a total loss of offsite power and special emergencies such as earthquakes or floods. Therefore, we have assumed that all accidents initiated by losses of offsite power and certain special emergencies which affect the site are coupled so that a potential core melt at one unit implies a potential core melt at the other. This is conservative since it is possible that offsite power might be lost at only one unit and not all site special emergencies would have the same effect on both units.

The two reactors at ANO are totally different and practically independent. One significant dependency is offsite power. Since the purpose of the sensitivity study is to address the two unit issue in general, i.e., at a generic 2-unit B&W site, we have assumed for this analysis that the 2 units are identical with the same remaining plant life and for the initiating events affecting both units, the units are mutually dependent. Also given a core melt at one unit but not the other, the unaffected unit will be assumed, as a precaution, to be shut down indefinitely. As for the impact analysis, none of the modifications affected both units so all impact costs were doubled.

The initiating events affecting either unit individually, but not both units, are S_2 , S_3 , T_2 , T_3 , T_4 , T_5 , Fire, Internal Flood, and Lightning; those affecting both units are T_1 , Seismic, External Flood, and Extreme Wind. Therefore the probability of core melt at the site is 2 times the probability of a 1-unit core melt, $pm(1CM)$ + the probability of a 2-unit core melt, $pm(2CM)$.

Other changes to the Value-Impact Analysis are:

$$I_5'(j) = \Delta p_m(j)[2I_{51}(j) + 2I_{52}(j) + 1.5 * I_{53}(j)]$$

$$\text{and } V_1 = 50600 * 22 * [\Delta p_m(j|1CM) + 2 * \Delta p_m(j|2CM)]$$

$$\text{where } \Delta p_m(j) = \Delta p_m(j|1CM) + \Delta p_m(j|2CM)$$

Given core melt at two reactors on a site, the radiological source term could double from what we have initially assumed. This will give rise to some increase in the offsite 50-mile population dose. But, as discussed in Reference 35, the 50-mile population dose is relatively insensitive to the source term so these consequences will not double. Based upon an extrapolation of the information in Reference 35, we have assumed, given a two unit melt, that the offsite dose is a factor of 1.2 times that for the single unit melt. The onsite dose received by the operators will be assumed to double for the dual melts as noted above.

It was assumed that two add-on dedicated residual heat removal systems would have to be installed in order to fully protect both units. The cost therefore doubles from what is estimated in Appendix J.

11.4.2 Value-Impact Estimates for Two-Unit Melts

By making the assumptions described in the last section, estimates of the V-I ratio, net benefit, and dollars per person-rem for the two alternatives were made. This was done without revising the fault trees or re-solving the accident sequences. The results are tabulated in Table 11.2. The total

TABLE 11.2 ESTIMATE OF VALUE-IMPACT MEASURES FOR ACCIDENTS INVOLVING BOTH ANO UNITS

Alternative No.	Reduction in Core Melt Probability	Offsite Averted Dose (p-rem)	Total Averted Dose Ratio	V-I Analysis of Offsite Costs			V-I Analysis Based on Offsite and Onsite Costs		
				V-I Ratio	Net Benefit (\$ $\times 10^{-6}$)	Dollars per p-rem	V-I Ratio	Net Benefit (\$ $\times 10^{-6}$)	Dollars per p-rem
	Δp_m	V_2	ADR_n	VIR_0	NBV_0	DPR_0	VIR_n	NBV_n	DPR_n
1	1.59E-4	231	0.80	4.5E-3	- 30.6	133000	1.4E-2	- 21.9	44000
2	2.29E-4	266	0.88	1.3E-3	-118.6	446000	3.2E-3	-106.1	181000

"site" core melt probability was estimated to be approximately $2.47\text{E-}4$ per reactor year.

Some of the results for the 2-unit case are summarized below:

	p(1CM)	+	p(2CM)	=	p(CM at site)
BASE CASE	$1.39\text{E-}4$		$1.08\text{E-}4$		$2.47\text{E-}4$
ALT 1	$7.12\text{E-}5$		$1.67\text{E-}5$		$8.79\text{E-}5$
ALT 2	$7.26\text{E-}6$		$1.08\text{E-}5$		$1.81\text{E-}5$

and,	50-mile Pop. Dose (1CM)	+	50-mile Pop. Dose (2CM)	=	50-mile Pop. Dose (Site)
BASE CASE	7.2		8.9		16.1
ALT 1	4.1		1.5		5.6
ALT 2	3.1		0.9		4.0

all in person-rem.

It should be noted that the value-impact measures presented in Table 11.2 are very rough estimates which should only be used to gauge what would happen for two identical B&W reactors at the ANO site. For more precise estimates, a more detailed modeling of the systems interactions between units would be necessary as well as a PRA in the ANO-2 reactor.

In summary, two general observations can be made. First, the modifications which have the highest potential for reducing the "site" risk or core melt probability for the lowest cost will be those which increase the redundancy or availability of systems shared between units. Second, while it is easy to imagine that the onsite costs such as power replacement, loss of investment, and cleanup could double for accidents involving two units, it is less clear how the avertable offsite costs might be affected. Given factors such as emergency response, land interdiction and decontamination, the overall, avertable offsite contribution to risk may not increase dramatically. More specifically, the value-impact measures for ANO-1 show that neither alternative is favorable. Furthermore, the results of the 2-unit analysis are even less favorable than the single unit analysis, i.e., compare Tables 11.1 and 11.2 dollars per person-rem.

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APPENDIX A
SYSTEMS DESCRIPTIONS FOR THE
ANO-1 NUCLEAR POWER PLANT

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1.0 INTRODUCTION

The purpose of this appendix is to describe each of the systems involved in the analysis presented in Appendix B. The fault trees used in the analysis were taken, with minor changes, from the IREP ANO-1 PRA. Similarly, most of the descriptive material was taken directly from the IREP ANO-1 PRA, Chapter 6.0 of Volume I. This reference is:

Interim Reliability Evaluation Program: Analysis of the Arkansas Nuclear One-Unit 1 Nuclear Power Plant, G. J. Kolb, Principal Investigator, NUREG/CR-2787, SAND82-0978, June 1982.

Additional detail can be found in Appendix B, Volume 2, of NUREG/CR-2787.

1.1 System Information

Information for the systems analysis was gathered from a number of sources during IREP. The Final Safety Analysis Report and Technical Specifications for ANO-1 were used. The plant was visited. Arkansas Power & Light (AP&L) also supplied complete operational and emergency procedures for ANO-1 as well as all the relevant piping and instrumentation diagrams, electrical one-line and control drawings, functional logic diagrams, and all the licensee event reports (LERs). Most importantly, AP&L personnel were available as needed for discussions, including full time participation during the systems analysis task of a former ANO-1 assistant superintendent of operations.

Most of the information collected during IREP was not readily available for TAP A-45 and some changes had been made to the plant subsequent to IREP. In addition TAP A-45 also considered external events and performed an impact analysis requiring estimates of potential plant modifications. This resulted in two plant visits and further telephone discussions with plant personnel. The IREP analysts assimilated information on each system including instrumentation and control and any operator actions affecting the system. Testing and maintenance of the system components was investigated, and the normal and emergency operation of the system were studied. In addition, a failure modes and effects analysis was conducted for each system in relation to its support system dependencies. These steps were undertaken to develop as complete an understanding of the system as possible.

A detailed fault tree for each front-line system was then constructed with the top event determined from the event tree analysis. Several systems have different success criteria, depending on the specific event tree application. In these cases, a top level fault tree was developed for each event tree failure definition. The system was separated into

pipng segments (wiring segments for electrical or actuation systems), and the top level tree was developed to the extent necessary to portray all the pipe segments whose failures were sufficient to fail the system for the given application. Local faults were analyzed within each pipe segment, and support system interfaces were identified. Some front line systems were not analyzed to this detail, e.g., the power conversion system.

Support system models were developed for all the interfaces identified by the front line system fault trees and for interfaces with other support systems. However, not all support systems were developed in full detail. For example, the instrument air system is needed to fill some charging tanks. It was assumed, however, that the tanks were full (or else shutdown is required), so that the instrument air system was not analyzed further. The overall support system study was similar to that described above for the front line systems.

The systems analysis resulted in a very detailed model for most of ANO-1 systems which not only logically described hardware faults but also included test and maintenance unavailabilities. In addition, possible operator error inputs to the models were analyzed.

1.2 General Assumptions

General assumptions for the analysis were of three classes: fault tree development and quantification, system fault postulation and consideration, and inclusion of operator action.

1.2.1 Fault Tree Development and Quantification

Two systems and portions of another were not modeled in detail. Main feedwater system fault trees were only developed in those areas which interfaced with other systems defined as front line or support systems. All other MFWS faults were grouped as local faults and loss of main feedwater generic industry data were applied as appropriate.

In addition, the ERV was not analyzed in detail. At the time of the study, the block valve at ANO-1 was closed. The failures of the safety valves to both open and close were examined and found to be independent of other systems. To quantify these failures, generic industry data were used.

Finally, portions of the emergency feedwater initiation and control (EFIC) system were not analyzed to the detail of other portions of the system.

1.2.2 System Fault Postulation and Consideration

Four general assumptions of this nature were made:

1. System fault events which could also be accident initiators (e.g., loss of offsite power) were explicitly included as appropriate in each system fault tree.
2. Passive failures which were accident initiators were included in each system fault tree. In addition, single passive failures which could fail the entire system were included.
3. Diversion flow paths were considered as potential system failure modes for fluid delivery systems. However, each potential diversion flow path was included on the fault tree only if it resulted in failure of the system and its likelihood was comparable to other system faults. That is, a possible diversion flow path had to have a diameter not less than one-third (approximately 10% of the area) of the diameter of the intended flow path, and probability discrimination was also used to determine whether or not to develop a diversion flow path.
4. Spurious control faults of components after successful initial operation were considered only in those cases where the component was expected to receive an additional signal during the course of the accident to readjust or change its operating state.

1.2.3 Inclusion of Operator Action

As mentioned above, operator actions were an integral part of the models of the systems. Three general assumptions were made regarding operator actions. The first two concern the first type of operator action discussed above, and the third is of the fail-to-restore type. The assumptions were:

1. Operator errors of commission which misposition valves or fail other components in response to the accident were included only for those components which are specifically identified in procedures as requiring operator manipulation.
2. Consideration of operator action as a successful operating mode for systems was done only in those cases where a written procedure for system operation exists which specifies the required operator actions. That is, operator recovery actions were not explicitly considered in the fault tree, but were treated following the screening calculations for accident sequence frequencies. "Verify" statements in procedures were treated as recovery operations.
3. Mispositioning of valves prior to the accident was not considered in those cases where valve position is indicated in the control room and monitored each shift. Nor was it considered if the valves receive an automatic signal to return to their operable state under accident conditions.

Most of the assumptions described above were made in IREP and in general carry over into TAP A-45; however, some details were not handled quite the same.

1.3 Fault Trees

Table A.1 gives a list of the fault trees used in the analysis and the appendix section where the system is described. Some systems have fault trees with more than one top event. Support systems are not shown as explicit fault trees but are only connected whenever applicable to the front line systems. Some main line systems have interrelationships such as the LPRS and HPRS.

1.4 Some TAP-A-45 Analysis Assumptions

A few additional assumptions which should be noted are:

1. Containment failure does not imply core melt due to containment sump flashing as has been assumed in many past PRAs. This permits the containment systems to be shown in a separate event tree.
2. The EFWS is sufficient to remove all decay heat without the RHR heat exchangers during the shutdown heat removal phase, i.e., to hot shutdown.
3. Any valves and pumps that were actuated for injection will still need actuation and power for the recirculation phase in case the injection was not required or pumps were turned off prior to the recirculation phase.
4. In order to avoid a fault tree loop where the diesel generator depends on the SWS which, in turn, depends on the diesel generator, it is assumed that only component faults of the applicable SWS train result in failure of corresponding diesel generator.
5. Two service water system pumps are normally running. Even though it can be any two, two specific pumps were selected for the model.
6. Reactor coolant pump seal failure is not considered since this is an issue which, if determined to be important, will be solved by another program.
7. In the event L defined as the EFWS and secondary steam relief, the secondary steam relief is not fault tree modeled since there is adequate redundancy, i.e., ADVs, or safety relief valves, or the steam dump to the condenser given condenser vacuum is available except in the case of a double steam generator tube rupture which is not being addressed.

Table A.1 System Top Event Names

<u>Section</u>	<u>Systems</u>	<u>Top Event Name</u>	<u>Event Tree/Accident Sequence Symbol</u>
2.0	EFWS	EFWF-10F2P	L
3.0	HPIS	HPIF-10F3P	D ₁
	HPRS	HPRF-10F3P	H ₁
	FNB	HPIF-10F3P-MAN	E
4.0	LPIS	LPIF-10F2P	D ₂
	LPRS	LPRF-10F2P-NHX	H ₂ '
5.0	RBSI	RBSIF-10F2P	C
	RBSR	RBSRF-10F2P	F
6.0	PRV	SRV-ERV-FAIL	P ₁
7.0	RBCS	RBCF-10F4U	Y
8.0	ESAS	LOS-UC1-LOG . . . LOS-UC8-LOG	
9.0	PCS	No Fault Trees	M
10.0	SWAC	FOP-OAC-SWS-S1 FOP-OAC-SWS-S2 FOP-OAC-SWS-S50	
	SWAL	FOP-OUT-SWS-S28 FOP-OUT-SWS-S29 FOP-OUT-SWS-S30 FOP-OUT-SWS-S31 FOP-OUT-SWS-S32 FOP-OUT-SWS-S36 FOP-OUT-SWS-S76 FOP-OUT-SWS-S77 FOP-OUT-SWS-S78 FOP-OUT-SWS-S79 FOP-OUT-SWS-S82 FOP-OUT-SWS-S83 FOP-OUT-SWS-S713 FOP-OUT-SWS-S728	

Table A.1 System Top Event Names (cont.)

<u>Section</u>	<u>Systems</u>	<u>Top Event Name</u>	<u>Event Tree/Accident Sequence Symbol</u>
11.0	EPS	B52-480VAC-LPW B62-480VAC-LPW RA1-125VDC-LPW RA2-125VDC-LPW D11-125VDC-LPW D21-125VDC-LPW RS1-120VDC-LPW RS2-120VDC-LPW RS3-120VDC-LPW RS4-120VDC-LPW A3-4160VAC-LPW A4-4160VAC-LPW B51-480VAC-LPW B61-480VAC-LPW	

2.0 EMERGENCY FEEDWATER SYSTEM (EFWS)

The purpose of the EFWS is to backup the main feedwater system (MFWS) in removing post-shutdown decay heat from the reactor coolant system via the steam generators. During normal shutdowns the MFWS is throttled down to a level capable of removing decay heat and the EFWS is not utilized. However, if the plant shutdown is caused by a loss of the MFWS or the reactor coolant pumps, or if the MFWS is lost subsequent to the plant shutdown, then the EFW system is put into operation. It is important to note that at some other PWRs the MFWS is not throttled down during normal shutdowns as at ANO-1. Instead, the MFWS is tripped and the backup feedwater system at these plants, the "auxiliary" feedwater system, is put into operation during all shutdowns. This note is made to explain why the backup feedwater system at ANO-1 is labeled emergency rather than auxiliary.

The EFW system consists of two interconnected trains, capable of supplying emergency feedwater (EFW) to either or both steam generators (SGs) from either of two water sources under automatic or manual initiation and control. A simplified piping diagram is included as Figure A.1. The system pumps take suction from either the condensate storage tank (CST) or from the service water system and discharge to the SGs. In the flow path between the EFW pumps and the SGs there are isolation valves, check valves, control valves, flow instrumentation, and pressure instrumentation to control the flow of EFW to the SGs. The EFW system is designed to provide a minimum of 500 gpm of EFW to the SGs at 1050 psig within 50 seconds of a system initiation signal.

Train A contains a motor-driven pump, and the pump of train B is turbine-driven. Except for electric motive and control power and actuation signals, the pumps, pump motor, and turbine are self-contained entities without support system dependencies. If AC power is not available, the B train can still provide complete system function relying solely on DC power.

It is assumed that the EFW pumps would fail by cavitation if the CST source was unavailable before the operator could realign the suctions of the pump to service water.

The success criteria of the system is to remove reactor coolant system decay heat from one of two steam generators. Either pump can supply sufficient feedwater for this purpose to either steam generator.

Several changes have been made to the EFWS since the IREP ANO-1 analysis. These changes have been incorporated into the model and are listed below for reference (see also Figure A.2 for the IREP schematic):

A-10

[illegible]

Figure A.1 Pipe Segments for the Emergency Feedwater System (TAP A-45)

1. One of the two motor operated valves in the steam line from steam generator E24A to the turbine driven pump has been closed and power disconnected.
2. The four normally closed discharge motor operated valves are now normally open.
3. The two pneumatic operated valves in the turbine steam supply line have been removed.
4. The three relief valves in the turbine steam supply line have been removed, eliminating these as potential diversion paths.
5. The crossover line and valves (CV2813 & CV2814) have been removed.
6. A parallel pair of check valves (CS261 & CS262) have been added to the CST drop line.

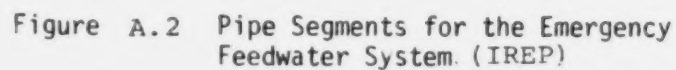
3.0 HIGH PRESSURE INJECTION AND RECIRCULATION SYSTEMS (HPIS & HPRS)

The high pressure (HP) system is utilized during those LOCAs where the reactor coolant pressure remains high (i.e., above about 150 psig where the low pressure (LP) pumps are ineffective). This condition will typically exist during small breaks and during the early stages of medium breaks. The high pressure injection system is, like most other engineered safeguards (ES) systems, actuated upon receiving an engineered safeguards actuation system (ESAS) signal which signifies either a 1500 psig RCS pressure or a 4 psig reactor building pressure. During the injection mode, the HP system draws borated water from the borated water storage tank (BWST) via a common tank outlet header shared with the LP and reactor building spray (RBS) systems. When switched to the recirculation mode (which requires manual operator actions) the water is drawn from the reactor building sump by the LP pumps through the decay heat coolers whose discharge is aligned to the HP pump suction which then injects into the reactor vessel. Figures A.3 and A.4 are a simplified schematic of the HP system (the discharges of the decay heat coolers to the HP pumps is through pipe segments DH7A and DH7B) with valve positions shown prior to injection. Room and lube oil cooling pipe segments are shown in Figure A.5.

The HP System is a two train, three pump system which injects water into the reactor pressure vessel via four injection headers (one for each cold leg of the RCS). The injection headers are cross connected such that each pump has an open flow path to all four RCS cold legs.

(Text continued on Page 16.)

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Figure A.3 Pipe Segments of High Pressure Injection System.

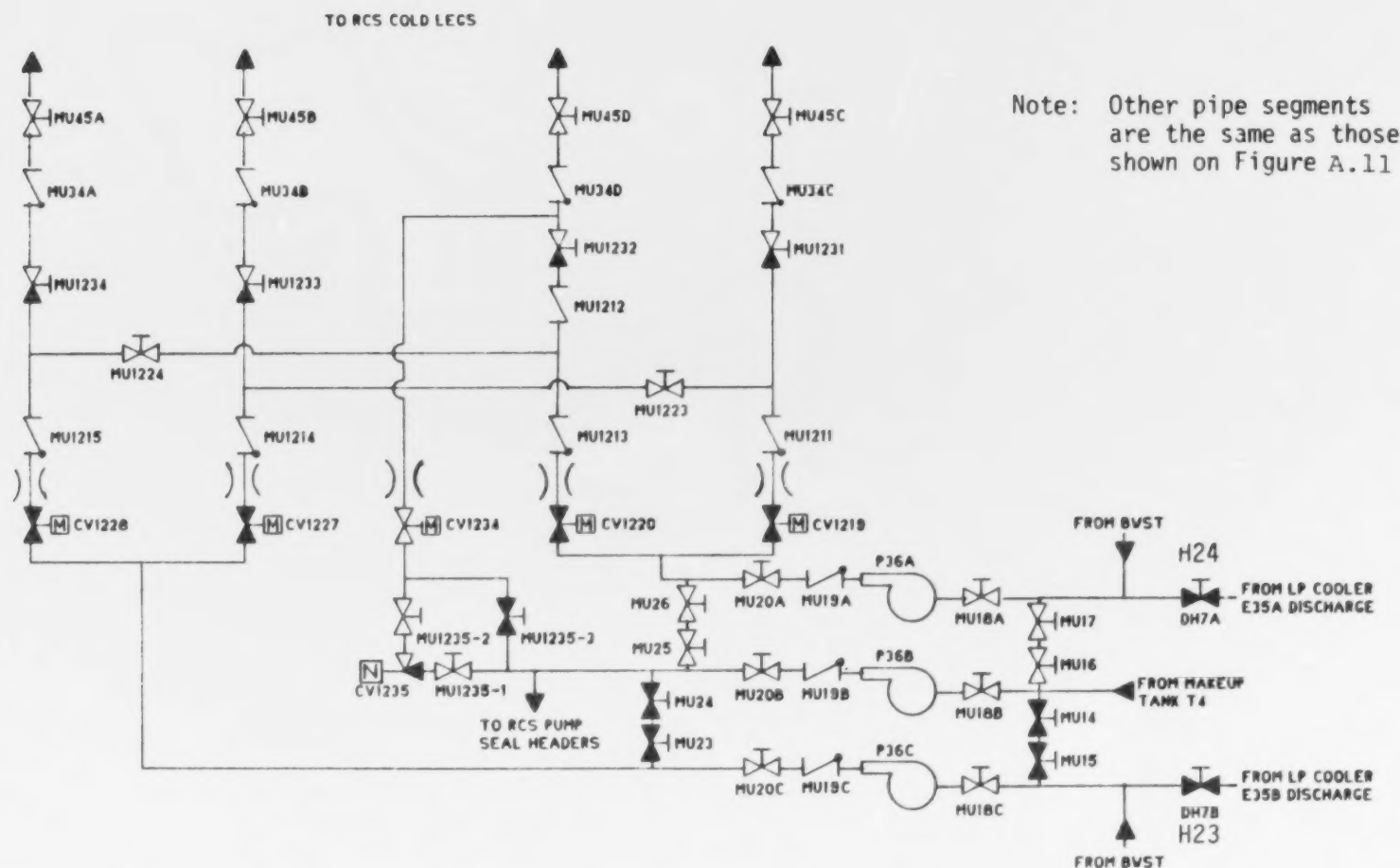


Figure A.4 Pipe Segments for High Pressure Recirculation System.

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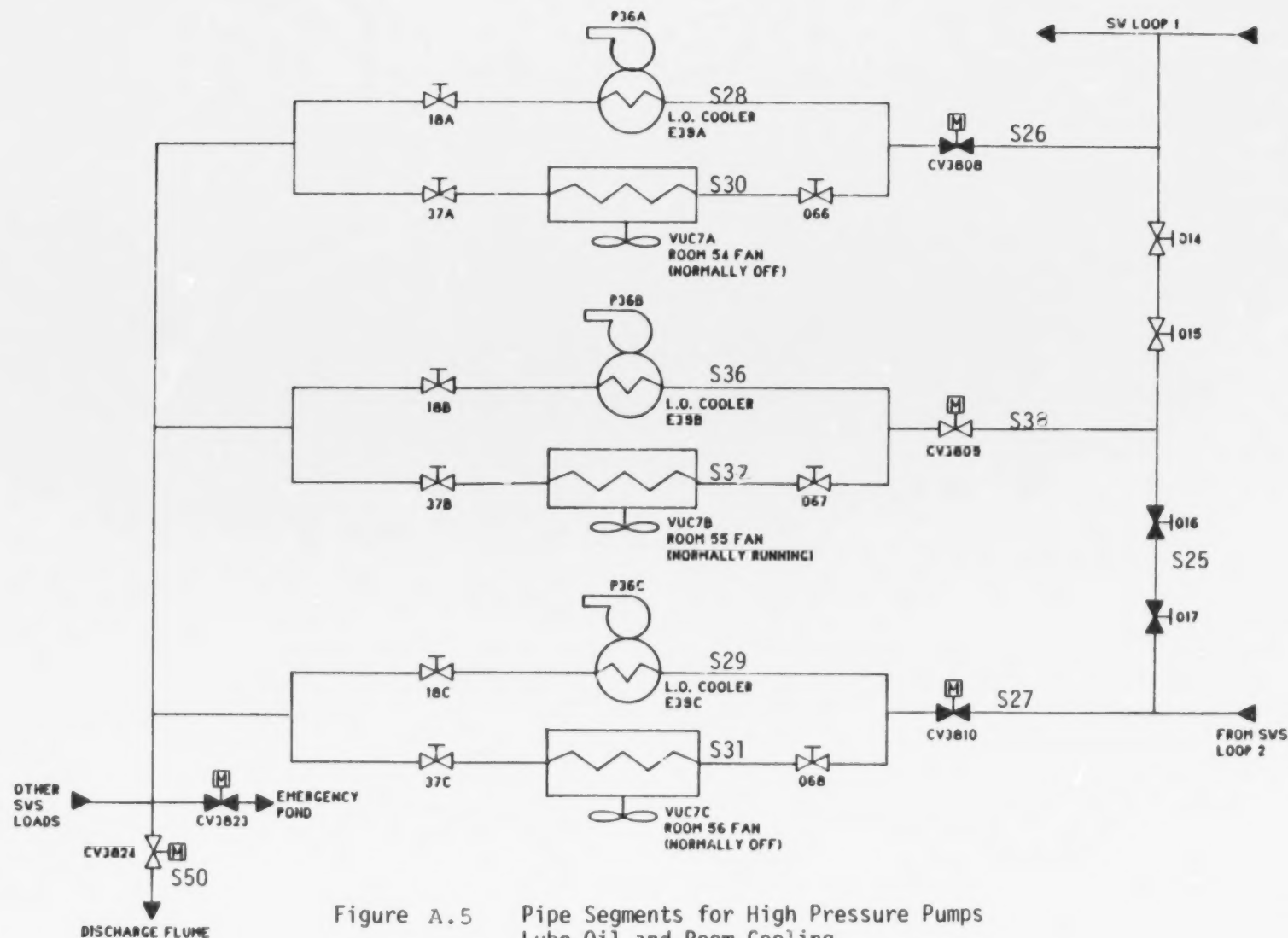


Figure A.5 Pipe Segments for High Pressure Pumps Lube Oil and Room Cooling.

During normal operation, one of three HP pumps is kept running in order to provide normal makeup to the reactor coolant system. Upon receiving an ESAS signal a second HP pump is started and the running HP pump is realigned from normal makeup to HPI. The realignment is accomplished by opening the suction of the HP pumps to the BWST, isolating the normal makeup (MU) tank and by realigning the discharge from the normal MU to the HPI piping. Although there are three pumps in the HP system, only two can be run at any one time since there are only two electrical power source busses and each can power only one pump. The A and C pumps are each connected to a different electrical bus and pump B can swing between the two busses. During normal operation, pump B is aligned to the bus powering the normal MU pump. The like-aligned pump is then configured as an automatic backup to the operating MU pump and the opposite-aligned pump is the ES pump.

Another function of the HPIS is to provide cooling water to the reactor coolant pump seals. The service water system provides a backup method of seal cooling through the intermediate cooling water system (ICWS). The ICWS, however, is isolated upon an ESAS signal.

The success criteria for the system are dependent on the initiating event. For example, for small LOCA and feed and bleed only one pump is necessary.

An important system assumption is that the failure of decay heat coolers to cool the recirculation water during HPR will fail the HP pumps as they are designed to pump water which is below 200°F.

An insight gained from the IREP study is that without recovery single failures exist for the HP system. During the recirculation phase of a LOCA, the HP pumps are kept below their design operating temperature by a single room cooler. Two other nonoperating coolers could potentially be used, but they must be started by the operator outside the control room.

4.0 LOW PRESSURE INJECTION AND RECIRCULATION SYSTEMS (LPIS & LPRS)

The LPI/LPR system serves a number of functions during both accident conditions and normal operations. Along with providing decay heat removal and filling and draining of the fuel transfer canal during plant outages and refueling, the system also provides emergency core cooling (ECC) during a LOCA. Although the system is actuated by an ESAS signal (i.e., 1500 psig RCS pressure or 4 psig RB pressure), it is not until the reactor coolant system (RCS) pressure drops to about 150 psig (as during a large break LOCA) that the system is able to overcome the RCS pressure and inject borated water into the

RCS. In addition to LP recirculation from the reactor sump, the system is utilized during the recirculation phase of a small break LOCA (i.e., RCS pressure remains high) in the DHRS mode of recirculation. During HPRS recirculation, the system is required to feed cooled (via the decay heat coolers) water from the RB sump to the HP system for injection into the pressure vessel. The LPRS, in conjunction with the spray recirculation, also provides long-term cooling to the containment. A simplified drawing of the system is presented in Figures A.6 and A.7. The system dependencies of the low pressure system are shown in Figures A.8, A.9, and A.10.

The LPI/LPR System consists of two independent trains which draw water via a common header from the BWST. This header pipe also provides water to the HP and RBS systems. Each train of the LP system consists of a LP pump, a decay heat cooler, piping and valves. The water is injected directly into the pressure vessel through two LP injection headers which are cross-tied and contain flow restrictors such that each train injects 50% of its flow through each injection header. These injection headers are also used by the core flood system, with the tanks being isolated from the pressure vessel by two check valves, one in the LP injection line, one in the CFT header.

The system is realigned for recirculation manually by the operators. This is done by opening the motor operated isolation valves which allow flow from the RB sump to the LP pump suction. Flow is also manually aligned from the DH cooler discharge to the HP pump suction during HPR. This is accomplished by opening two manual valves (one in each train) in the cross-tie piping.

The success criteria of the LP injection system requires one of two pumps. In addition, during the recirculation phase of the accident sequences, one of two pumps is required.

For this system cavitation of the LP pumps due to the nonclosing of CV1407 and CV1408, when switching to recirculation, was assumed to be negligible in the analysis for two reasons. Both the BWST and RB sump are at higher elevations than the LP pumps, and the pumps are gravity fed at their suction, and secondly, discussions with ANO-1 personnel revealed that cavitation of the pumps was not of concern if the valves were left open.

5.0 REACTOR BUILDING SPRAY INJECTION AND RECIRCULATION SYSTEM (RBIS & RBRS)

For the purposes of this analysis, the Reactor Building spray system (RBSS) performs two functions. These functions are to (1) reduce the post-accident reactor building pressure to nearly atmospheric pressure during injection and act in

(Text continued on Page 23.)

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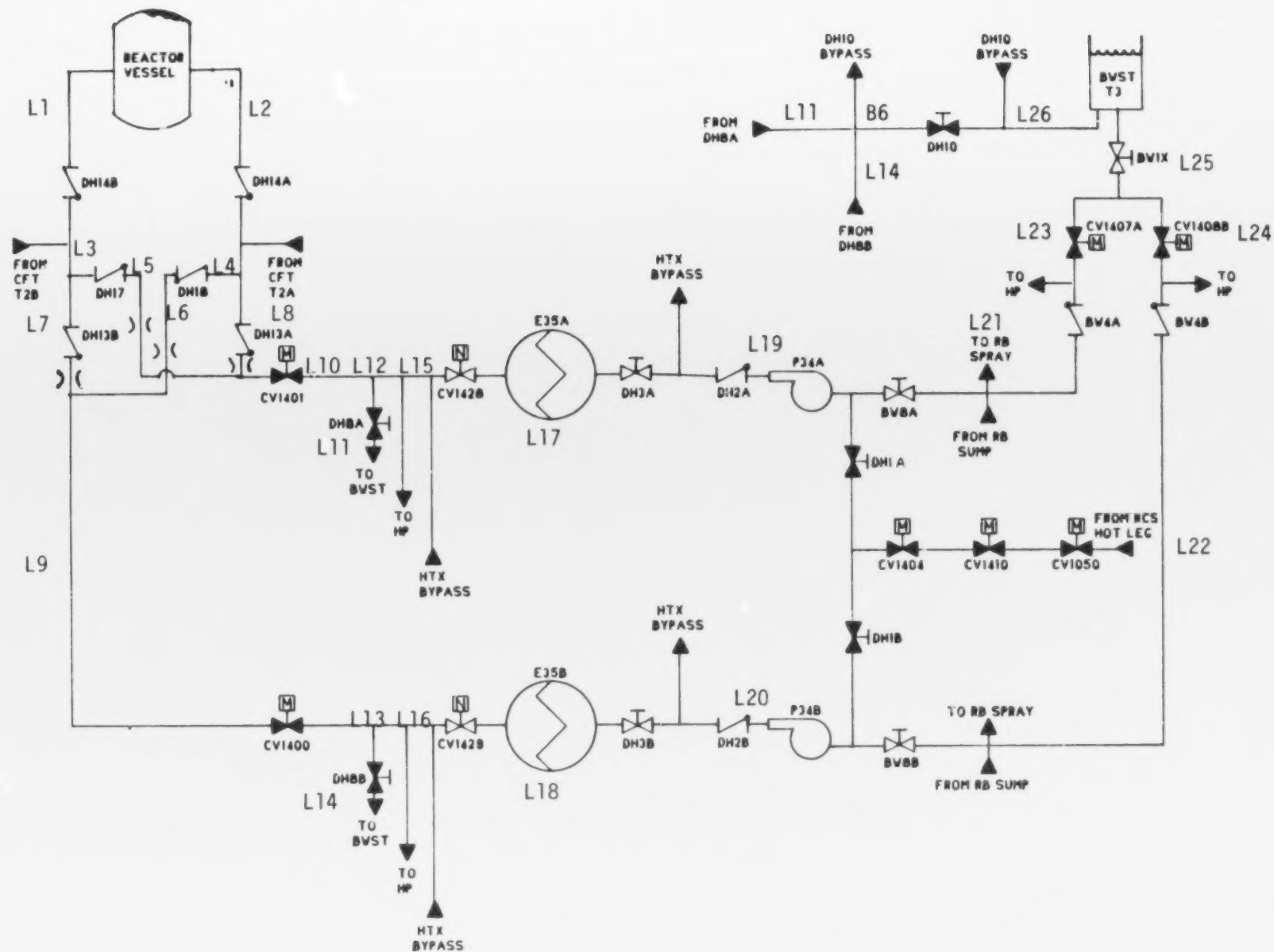
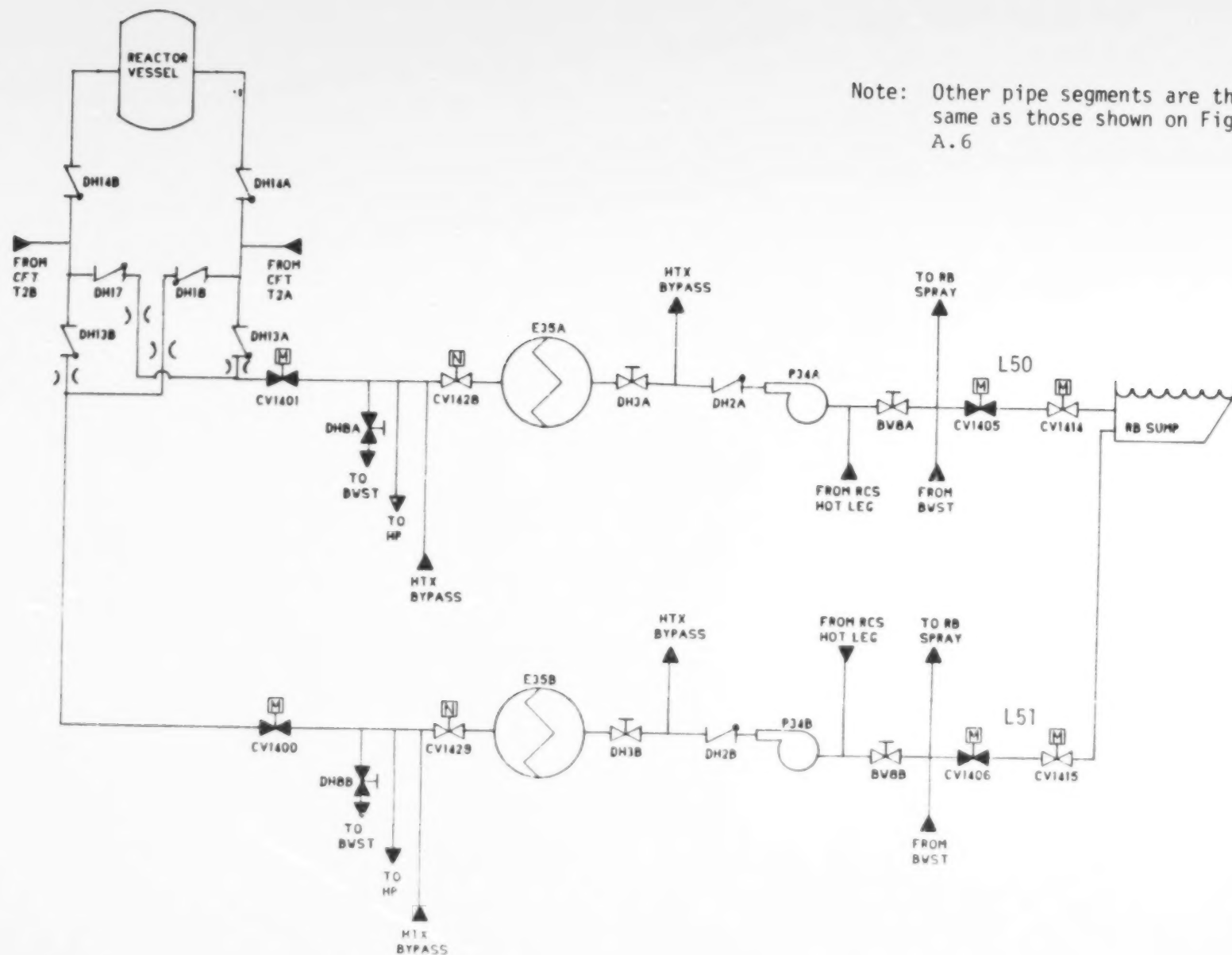


Figure A.6 Pipe Segments for the Low Pressure Injection System.

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Note: Other pipe segments are the same as those shown on Figure A.6

Figure A.7 Pipe Segments for the Low Pressure Recirculation System.

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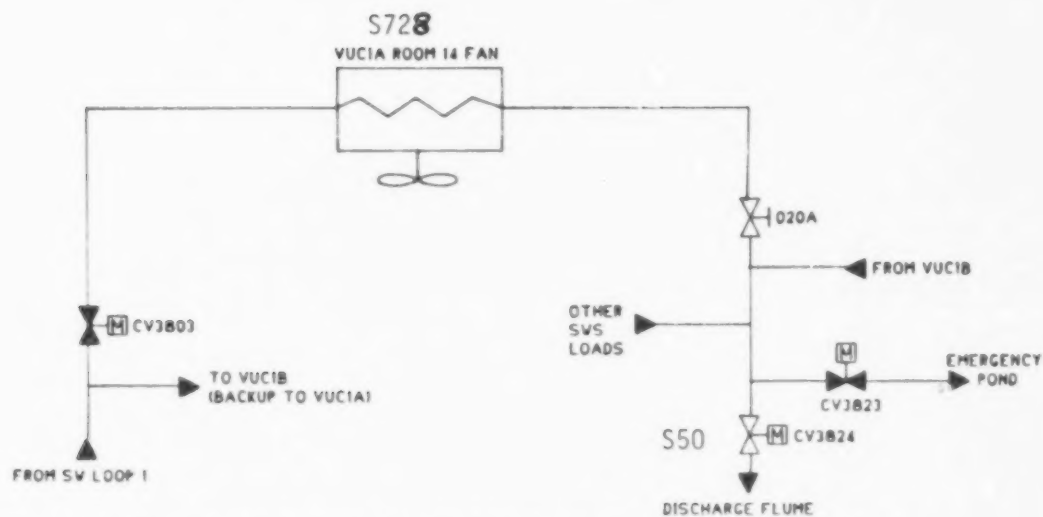
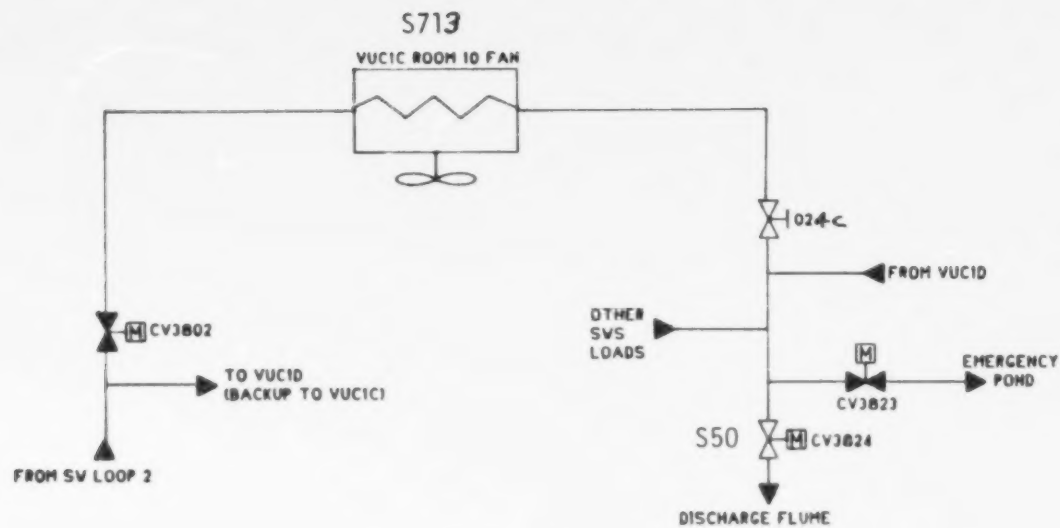


Figure A.8 Pipe Segments for Low Pressure Pump Room Coolers.

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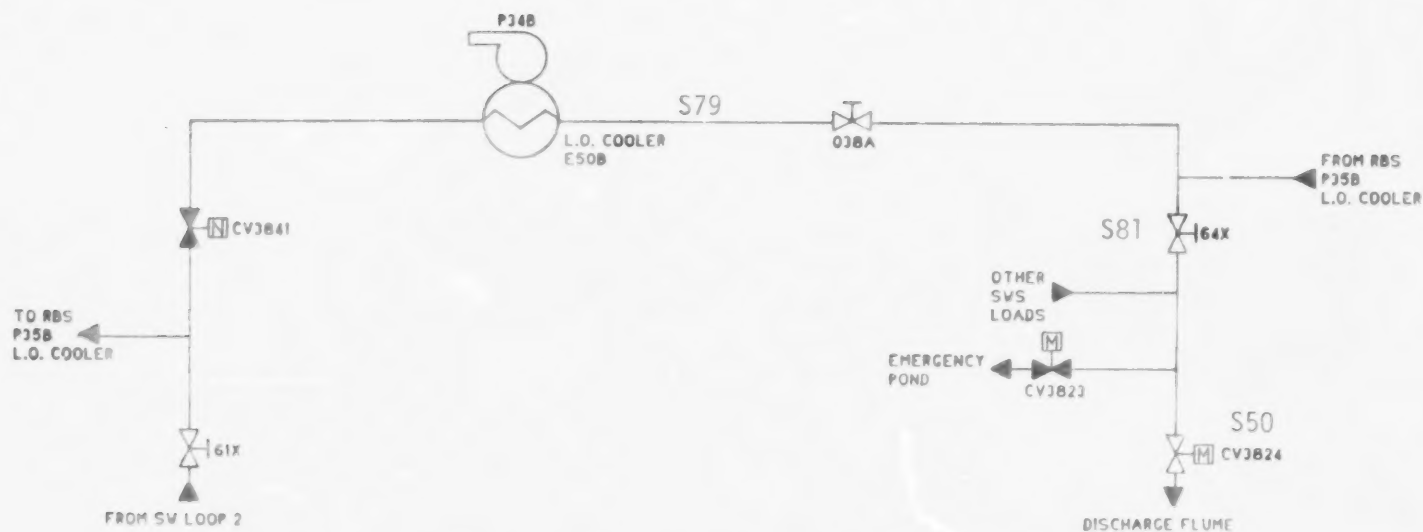
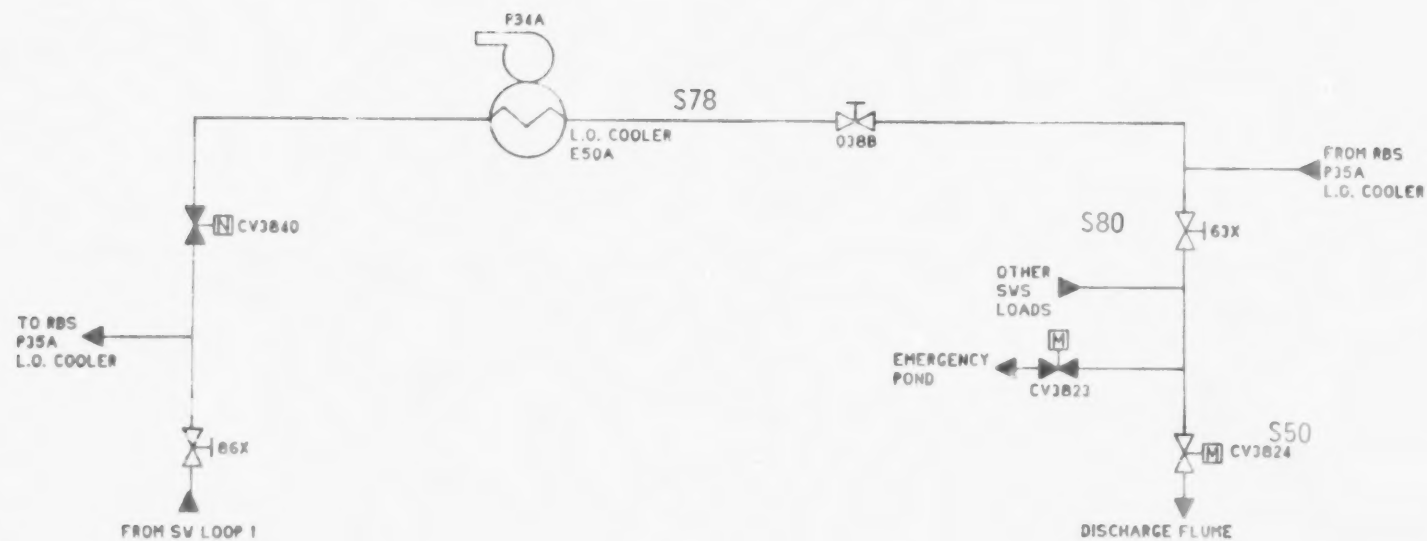


Figure A.9 Pipe Segments for the Low Pressure Pump Lube Oil Coolers.

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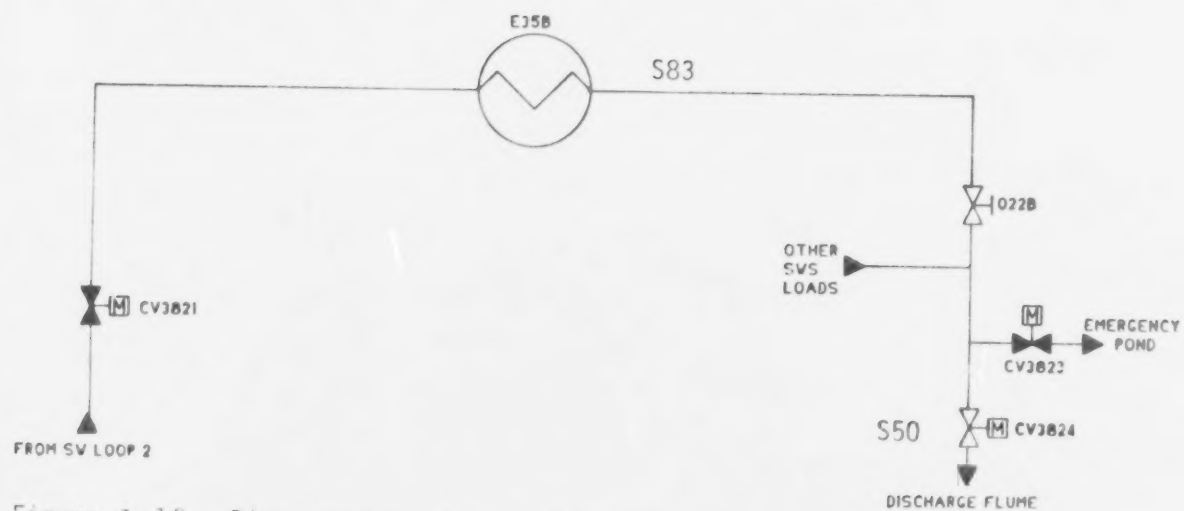
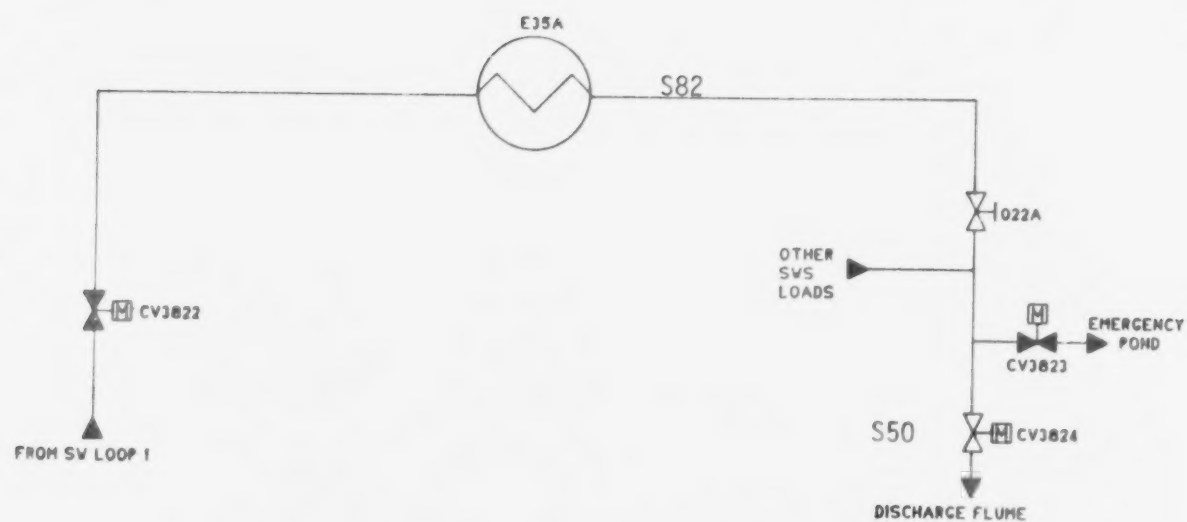


Figure A.10 Pipe Segments for Low Pressure Decay Heat Exchangers Cooling.

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conjunction with the low pressure recirculation system to remove heat from the containment during recirculation, and (2) remove radioactivity from the containment atmosphere.

The RBSS serves only as an engineered safeguard system and performs no normal operating function. It consists of two independent trains. In the event of a loss-of-coolant accident (LOCA) that results in a Reactor Building pressure of 30 psig, the RBSS is actuated by the ESAS and takes water from the low pressure injection lines that come from the borated water storage tank (BWST) and sprays it into the Reactor Building. Once the BWST reaches a low level, the RB spray pump suction is transferred to the Reactor Building sump. This first phase is called Reactor Building spray injection (RBSI) and the second phase is called Reactor Building spray recirculation (RBSR). A simplified schematic of the RBSS is shown in Figure A.11a.

The RBSS interfaces with the low pressure (LP) system in both the injection and recirculation phases. During the injection phase, in order for the water source to be available it is necessary that the outlet manual valve (BW-1) from the BWST and the motor operated valves (MOVs) (CV1407 and CV1408) in the low pressure line be open. (The manual valve BW-1 is normally locked open; however, MOVs CV1407 and CV1408 receive an ESAS to open on either of a 4 psig building pressure or 1500 psig coolant pressure. Thus, these valves should be open by the time the RBSS requires injection water.) The RBSS also requires LP valve action for recirculation.

The Reactor Building spray pumps require lube oil cooling. The flow paths modeled are shown in Figure A.11b.

For the two functions of the RBSS mentioned above, the success criteria is that one of the two trains must function. Two system-specific assumptions were made. First, the plugging of the spray nozzles was ignored. Second, the addition of sodium hydroxide to the spray was not modeled. The chemical addition to the spray does not significantly mitigate the offsite consequences of an accident.

6.0 PRESSURE CONTROL AND PROTECTION

Normal reactor coolant system pressure control is by the pressurizer steam cushion in conjunction with the pressurizer spray and heaters. The system is protected against overpressure by reactor protection system circuits such as the high pressure trip and by pressurizer safety valves located on the top head of the pressurizer. The schematic arrangement of the safety valves is shown in Figure A.12. Since all sources of heat in the system, i.e., core reactor coolant pumps, and pressurizer heaters, are interconnected by the reactor coolant piping with no intervening isolation valves, all safety valves

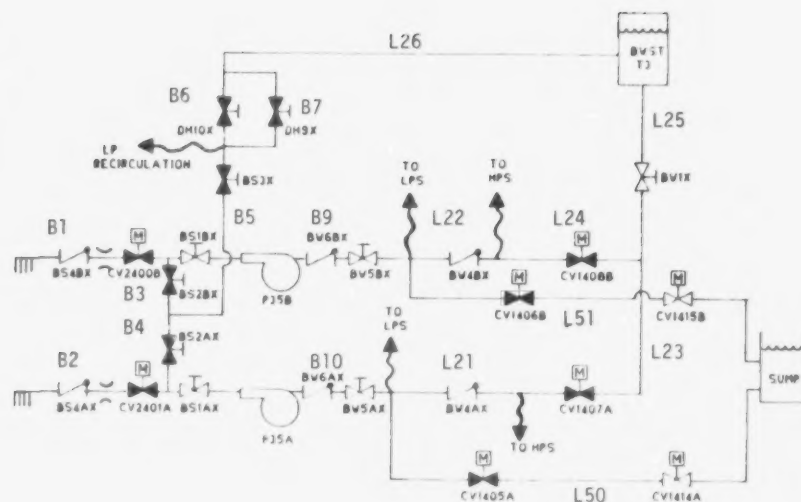


Figure A.11a Pipe Segments for the Reactor Building Spray Injection/Recirculation Systems

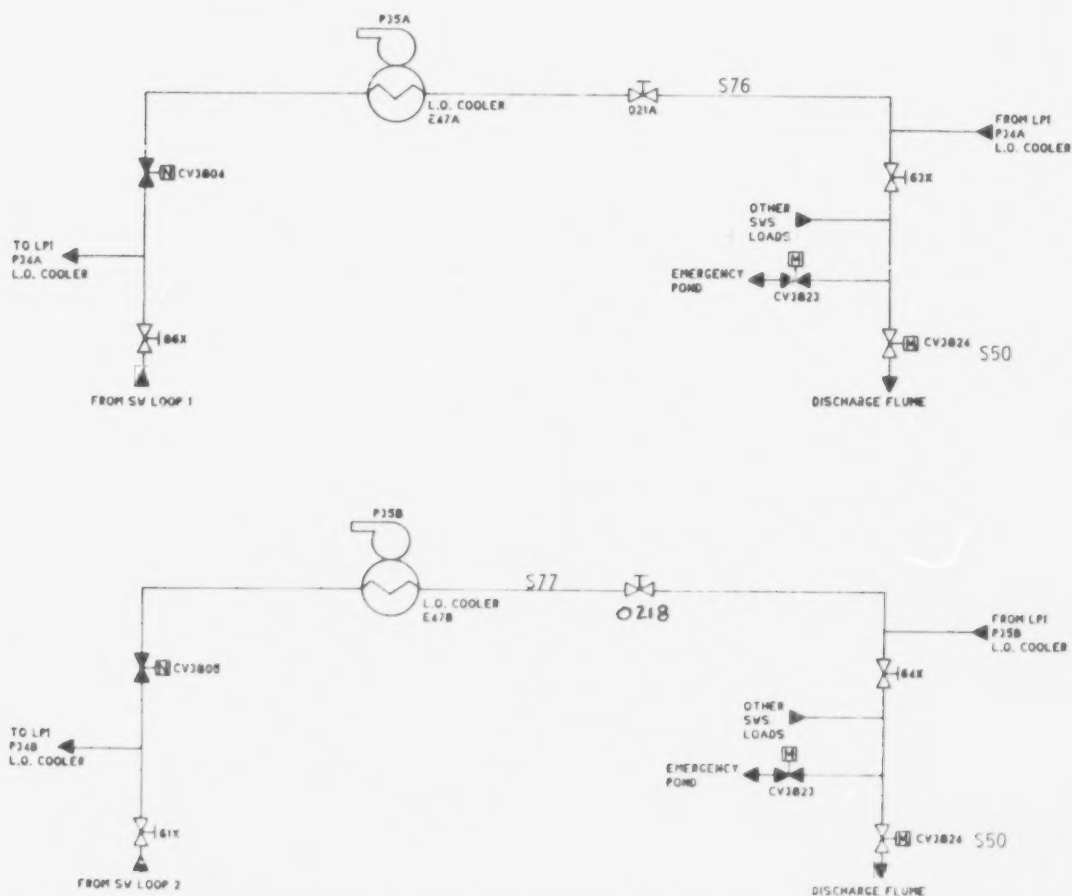


Figure A.11b Pipe Segments for Reactor Building Spray Pumps Lube Oil Cooling

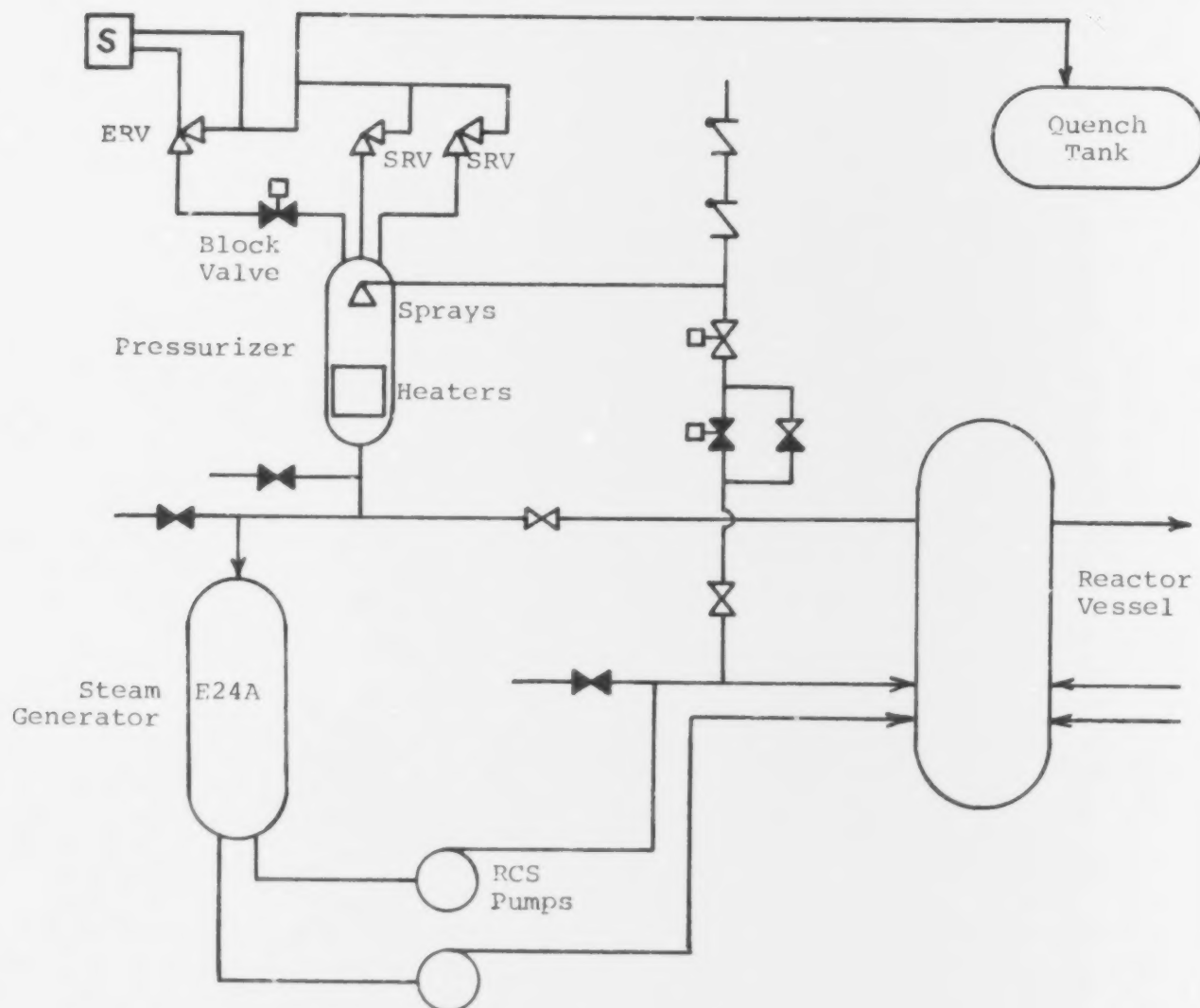


Figure A.12 Reactor Coolant System Pressurizer

are located on the pressurizer. Reactor coolant system pressure settings and safety valve capacities are listed in Table A.2. Most of the information presented in this section was taken directly from the ANO-1 FSAR.

Pressurizer Code Safety Valves - Two pressurizer code safety valves are mounted on individual nozzles on the top head of the pressurizer. The nozzles are designed to transfer full discharge loads to the pressurizer. The valves have a closed bonnet with bellows and supplementary balancing piston. The valve inlet and outlet are flanged to facilitate removal for maintenance or set point testing.

Pressurizer Electromatic Relief Valve - The pressurizer electromatic relief valve is mounted on a separate nozzle on the top head of the pressurizer. This nozzle is designed to transfer full discharge load to the pressurizer. The main valve operation is controlled by the opening or closing of a pilot valve which causes unbalanced forces to exist on the main valve disc. The pilot valve is opened or closed by a solenoid in response to the pressure set points. Flanged inlet and outlet connections provide ease of removal for maintenance purposes.

Pressurizer Spray - The pressurizer spray line originates at the discharge of a reactor coolant pump in the same heat transport loop that contains the pressurizer. Pressurizer spray flow is controlled by an electric motor operated valve using on-off control in response to the opening and closing pressure set points. An electric motor operated valve in series with the spray valve provides a backup means of securing flow in the event the spray valve should stick open.

Pressurizer Heaters - The pressurizer heaters replace heat lost during normal steady state operation, raise the pressure to normal operating pressure during reactor coolant system heatup from a cooled down condition, and restore system pressure following transients. The heaters are arranged in 14 groups and are controlled by the pressure controller. Two groups utilize modulating control, and will normally operate at partial capacity to replace heat lost, thus maintaining pressure at the set point. On-off control is used for the remaining 12 groups. A low level interlock prevents the heaters from being energized with the heaters uncovered.

Relief Valve Effluent - Effluent from the pressurizer electromatic-relief and code safety valves discharges into the quench tank, which condenses and collects the relief valve effluent. This is shown schematically in Figure A.12. The tank contents are cooled by dilution with condensate.

The quench tank is protected against overpressure by a rupture disc sized for the total combined relief capacity of the two

Table A.2 Reactor Coolant System Pressure Settings

	Pressure, <u>psig</u>	Design Capacity, <u>lb/h, total</u>
Design Pressure	2500	
Pressurizer Code Safety Valves	2500	600,000
High Pressure Reactor Trip*	** 2300	
Pressurizer Electromatic Relief Valve*		
Open	2455	
Closed	2420	100,000
High Pressure Alarm*	2255	
Pressurizer Spray Valve*		
Open	2205	
Close	2155	
Operating Pressure*	2155	
Low Pressure Alarm*	2055	
Low Low Pressure Alarm*	1920	
Low Pressure Reactor Trip*	1800	
Hydrotest Pressure	3125	

* At sensing nozzle on reactor outlet pipe

** There is a technical specification change amendment request into NRC to make this 2355.

pressurizer code safety valves and the pressurizer electromatic relief valve. The quench tank is vented to the gaseous radioactive waste disposal system.

SRV-ERV Fault Tree - The safety relief valve (SRV) - electromatic relief valve (ERV) fault tree was constructed for ANO using a TAP A-45 approach and is given in Figure A.13. The purpose of the block valve gate (BLK-VLV-GATE) is to permit failure of AC power to apply for cases where the ERV is blocked but to eliminate that failure event when the ERV is unblocked.

7.0 REACTOR BUILDING COOLING SYSTEM (RCLS)

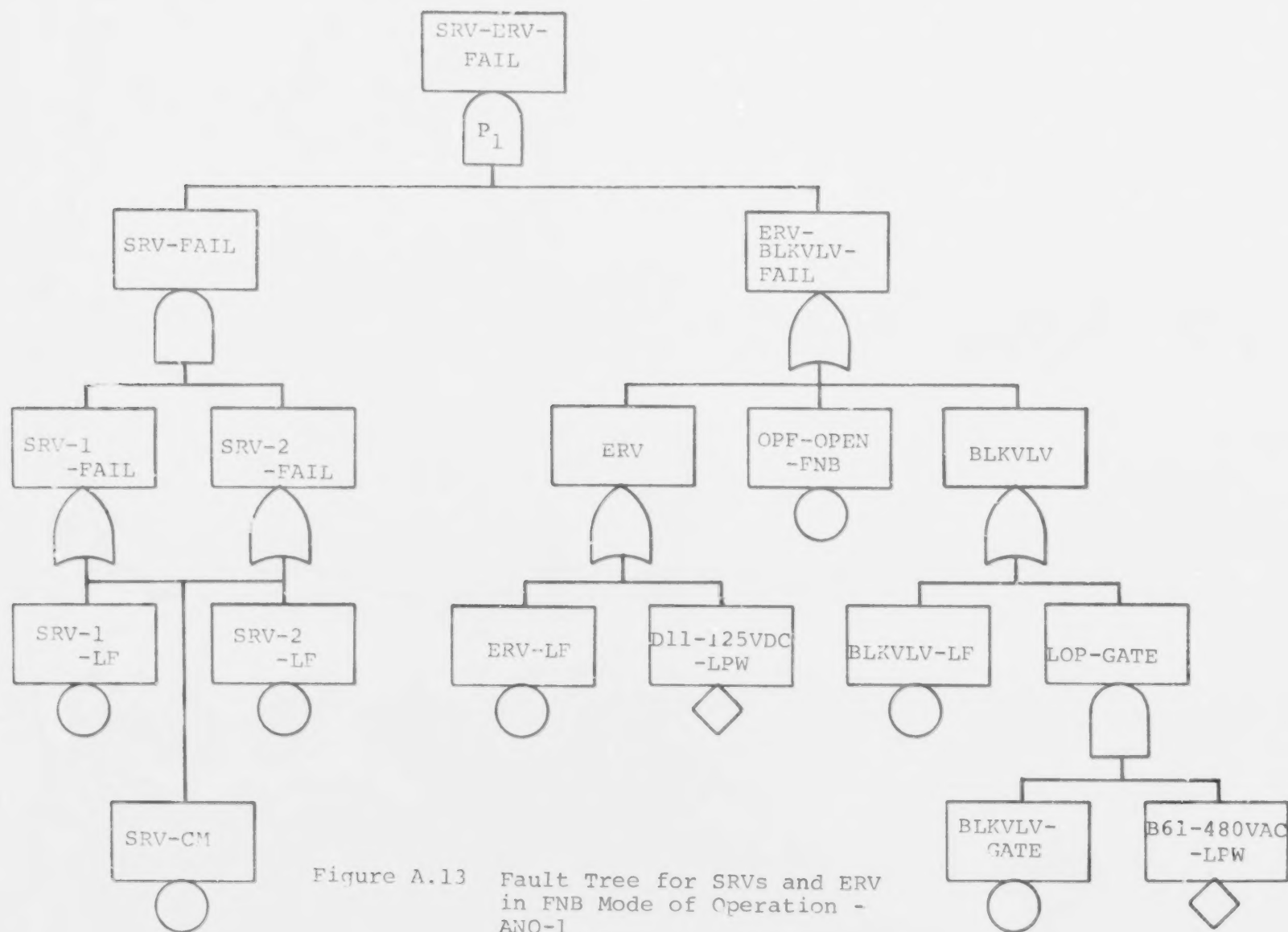
The Reactor Building cooling system (RBCS) is provided to limit post-accident Reactor Building pressure to the design value during steam evolution within the building due to an accident.

Emergency and normal cooling of the Reactor Building are performed with the same basic cooler units. Each unit contains normal and emergency cooling coils and a single speed fan. During normal plant operation, chilled water from the plant main water chillers is circulated through the normal cooling coils of the fans which are running. For emergency cooling all units operate under post-accident conditions with the heat being rejected to the service water system. ESAS-actuated dampers open to alter the flow from the normal to the emergency path.

The schematic flow diagram of the Reactor Building cooling system and associated instrumentation is shown in Figure A.14. The Reactor Building atmosphere enters each of the fan coolers at the fan locations. All four fans discharge into a supply air plenum which distributes cooled air throughout the Reactor Building. One pair of units (VSFM1A and VSFM1B) is cooled by service water loop 1 while the other pair (VSFM1C and VSFM1D) is cooled by service water loop 2. The Reactor Building is normally isolated from the service water system by two ESAS-actuated, pneumatically operated valves and two ESAS-actuated motor operated valves. A safeguards actuating signal, generated by 4 psig pressure in the Reactor Building, will cause the valves to open. After the service water exits the Reactor Building, it is monitored for high radiation. High radiation in the service water closes the service water isolation valves and overrides any existing ES signal.

As with the service water and actuation signals, the four fans are divided as to motive power. The A and B fans are supplied from the odd emergency AC train, and the C and D fans from the even. Similarly, the associated valving is powered from the odd or even AC train, respectively, or the odd or even DC power train.

The success criterion of the RBCS is that one of the four fan coolers must operate to fulfill the function of the RBCS.



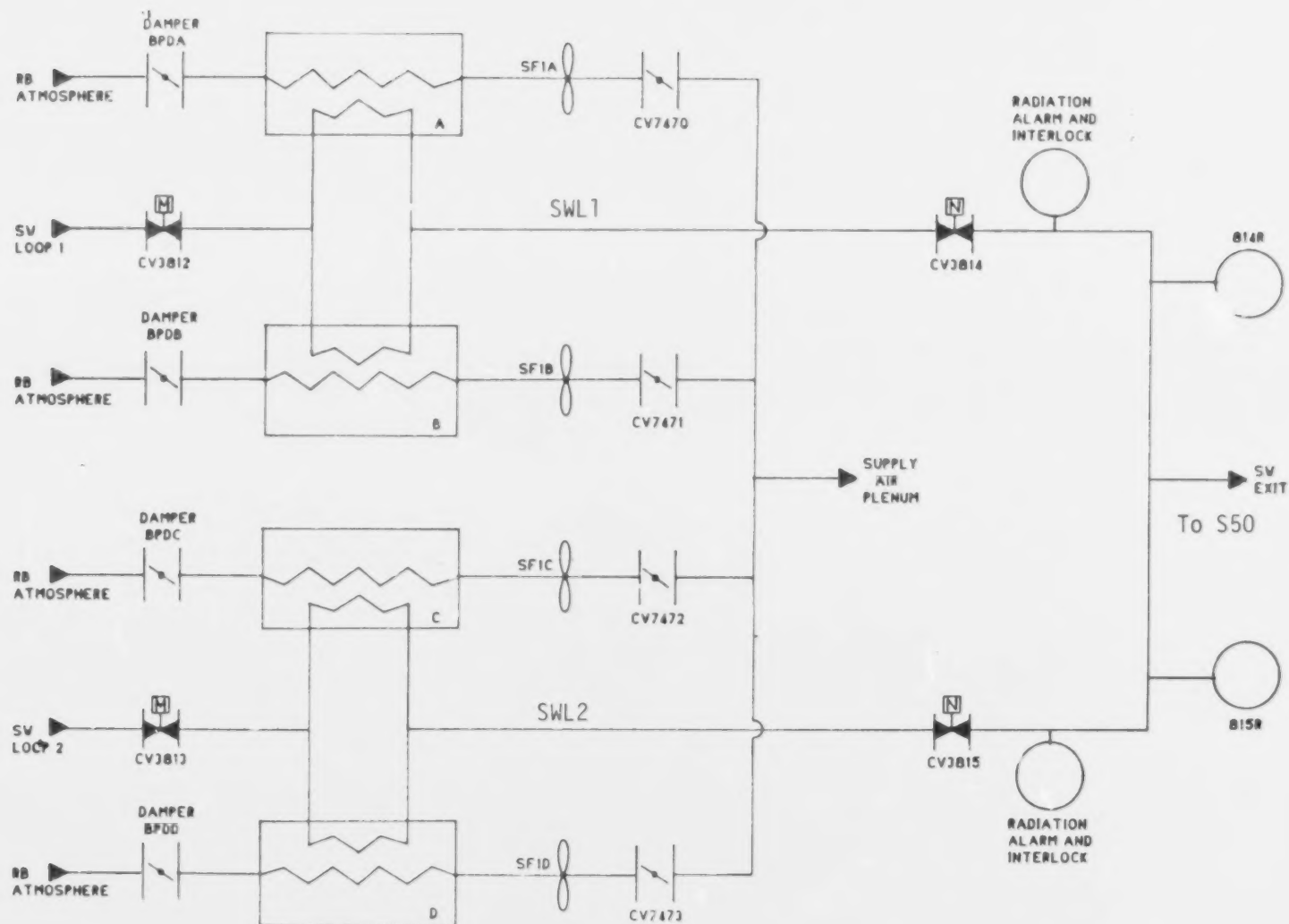


Figure A.14 Pipe Segments for the Reactor Building Cooling System

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8.0 ENGINEERED SAFEGUARDS ACTUATION SYSTEM (ESAS)

The engineered safeguards actuation system (ESAS) monitors parameters associated with a major loss of reactor coolant accident and initiates operation of the proper engineered safeguards systems, i.e., emergency core cooling, Reactor Building isolation and cooling, and Reactor Building spray, dependent on the severity of the accident.

In addition, ESAS starts the diesel generators but does not connect them to the emergency AC electrical system.

ESAS is composed of three redundant analog subsystems and two redundant digital subsystems as shown in Figure A.15. Each analog subsystem contains two channels which monitor reactor coolant pressure and Reactor Building pressure. Each of the two digital subsystems contains five logic channels for initiation of safety action when two of three analog subsystems indicate such action is required. The components actuated by the odd and even digital subsystems are generally different but complementary, e.g., the pump in an odd system train is actuated by the odd ESAS train and the even by the even.

Each analog train has two pressure sensors, one for the reactor coolant and one for the Reactor Building atmosphere. Each sensor is connected to a buffer amplifier, which in turn feeds at least two bistables. Trip points for the sensors are less than 1500 psig for the coolant and greater than 4 psig (high) or 30 psig (high-high) for the atmosphere, depending on which digital train is to be tripped. The analog components fail safe, i.e., removal of, or loss of power to, a module causes the train output to initiate, which results in a 1 of 2 logic configuration remaining. During normal shutdown the low reactor coolant pressure trip can be manually bypassed during the cooldown pressure interval of 1750 to 1500 psig.

Each channel can be tripped by 1 of 2 methods: 2 of 3 analog subsystems trip, or manual action. Removal of, or loss of power to, a digital component module does not cause an actuation output signal. The modules in the digital portion can be tested on-line. The success criteria of the eight logic trains is that each must send its signal to its actuated components when the specific accident sequence requires it.

More detailed system block diagrams are given in Figures A.16 and A.17 for the eight actuation channels used in the analysis.

9.0 POWER CONVERSION SYSTEM (PCS)

The power conversion system (PCS) at ANO-1 is designed to provide feedwater to the secondary side of the steam generators which, in turn, transfer energy to the turbine generator system. Following a reactor trip, the PCS is also capable of delivering

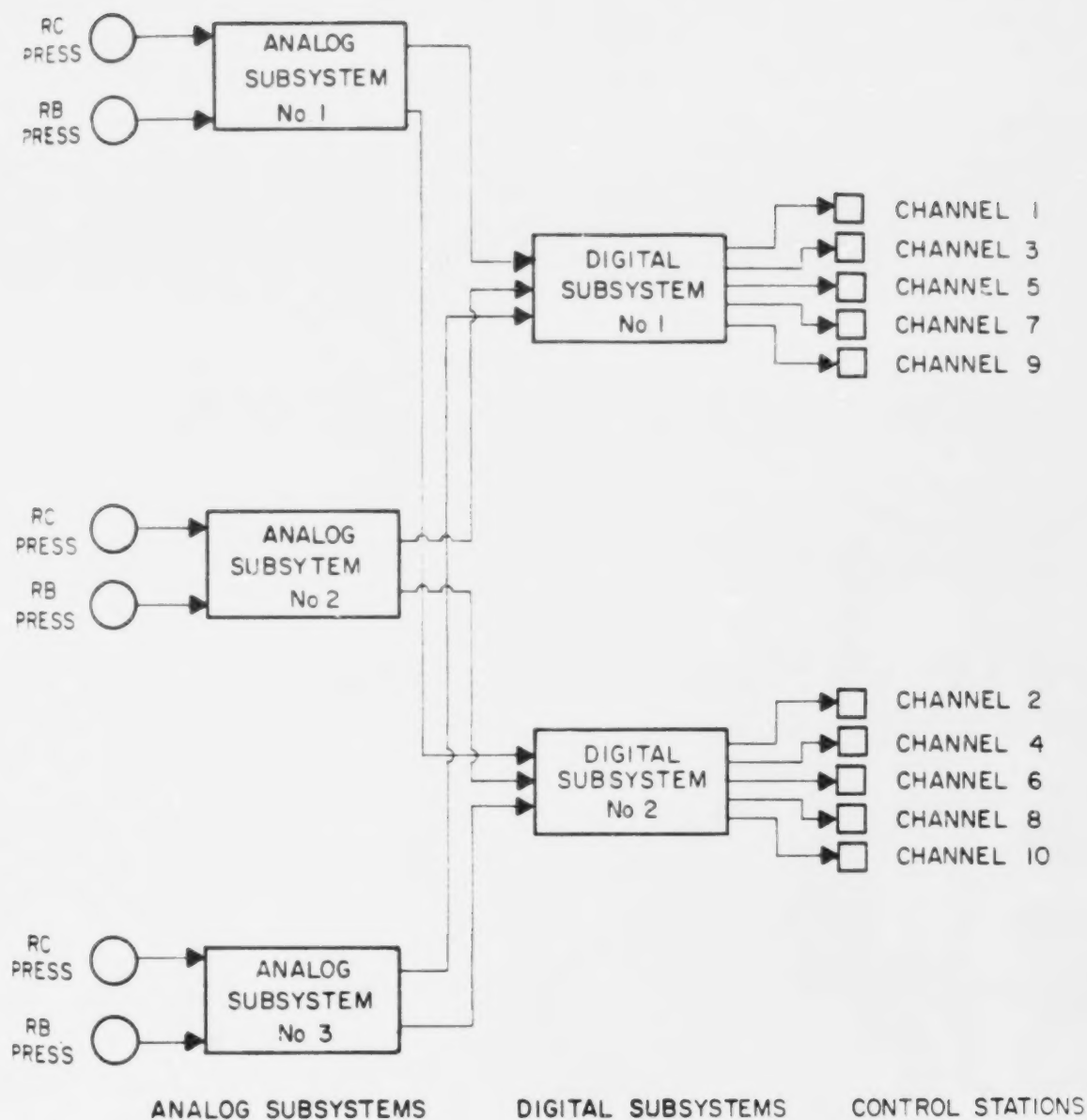


Figure A.15 Simplified Schematic of Engineered Safeguards Actuation System.

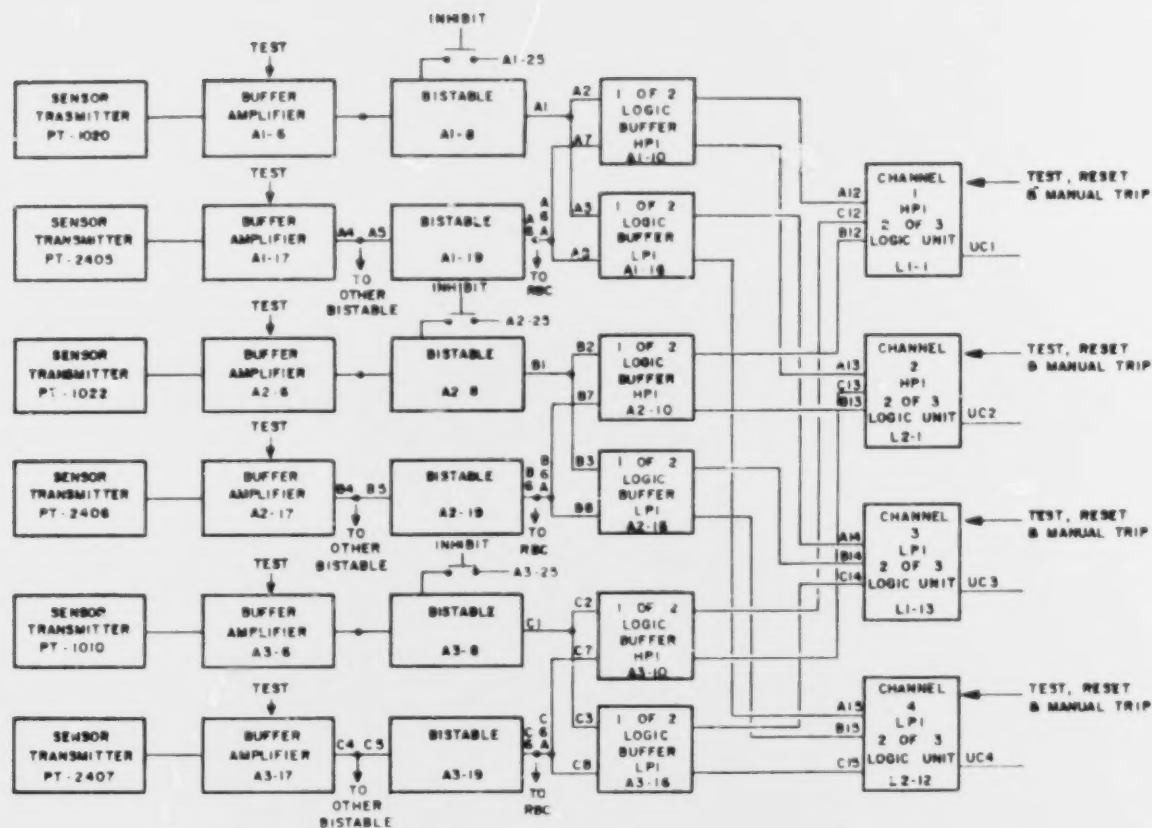


Figure A.16 Wiring Segments of Channels 1-4 of the Engineered Safeguards Actuation System.

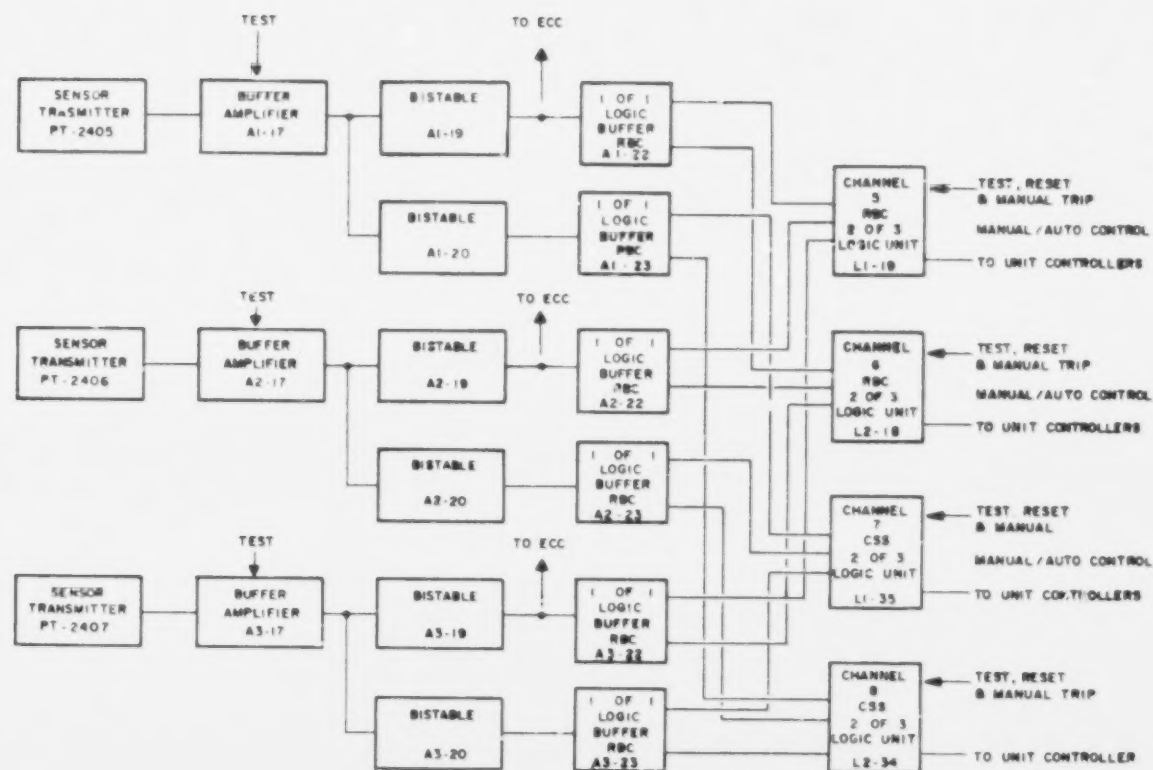


Figure A-17 Block Diagram of Channels 5-8 of the Engineered Safeguards Actuation System.

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feedwater to the steam generators at a reduced rate to provide for decay heat removal. This is accomplished by throttling the PCS feedwater flow to a level commensurate with decay heat and allowing this water to boil off to the condenser or atmosphere.

Figure A.18 shows a simplified schematic of the PCS. The feedwater portion of the PCS consists of two pump trains. Three low pressure motor driven condensate pumps feed two high pressure steam driven main feedwater pumps and one high pressure motor driven auxiliary feedwater pump. These latter three pumps in turn feed both steam generators via two injection lines. The injection lines are interconnected by a line which contains a motor operated valve. Each injection line also includes three parallel lines, two with control valves for startup and low load modulation and one with a motorized gate valve for full load; one high pressure feedwater heater; one flow tube type flow measuring device; isolation valves; and one check valve (containment isolation valve) just outside the reactor containment.

A detailed fault tree of the PCS was not constructed. Instead, industry data were used which represent the PCS unavailability due to independent causes not associated with the initiating events or systems modeled in the fault trees. Also, in IREP a review of all support systems interfacing with the PCS was made to determine which component failures already modeled in the fault trees would cause total interruption of the system.

10.0 SERVICE WATER SYSTEM (SWS)

The purpose of the service water system (SWS) is to provide cooling water for the following equipment during emergency conditions:

1. Reactor Building cooling system cooling coils,
2. Diesel generator jacket heat exchangers,
3. High pressure pump lube oil coolers,
4. High pressure pump room coolers,
5. Circulating water pumps bearing lubrication,
6. Low pressure/building spray pump room coolers,
7. Low pressure pump(s) bearing coolers,
8. Low pressure system heat exchangers,
9. Building spray pump(s) bearing L.O. coolers,
10. Emergency feedwater system water sources.

The SWS consists of two redundant loops as shown in Figure A.19. Normal cooling is supplied from Lake Dardanelle; however, an emergency pond is available in case of loss of flow from the lake. The service water is normally discharged back to the lake via the circulating water discharge flume. If the lake source is lost, the service water would be discharged back to the emergency pond.

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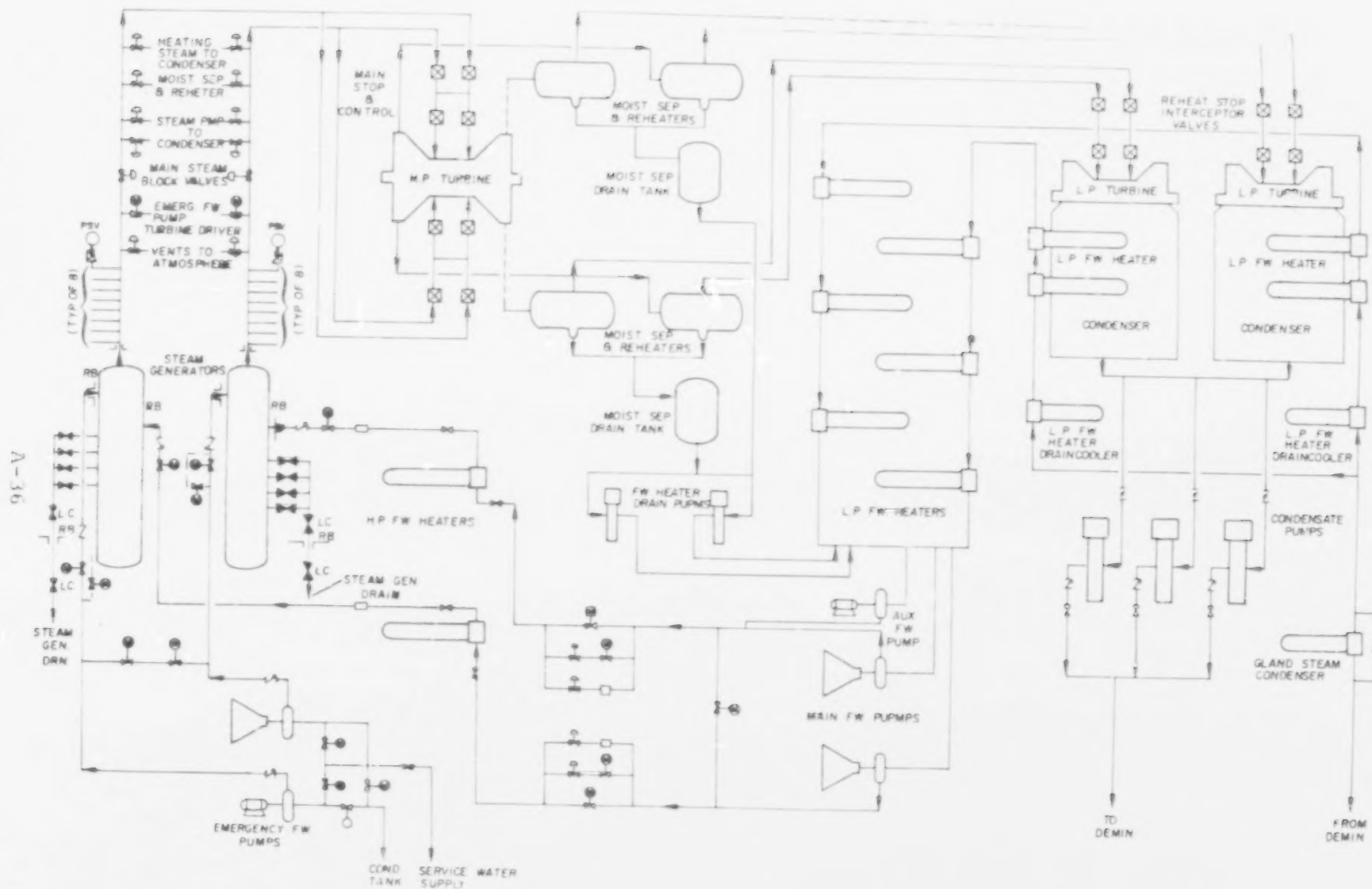


Figure A.18 Schematic of Power Conversion System

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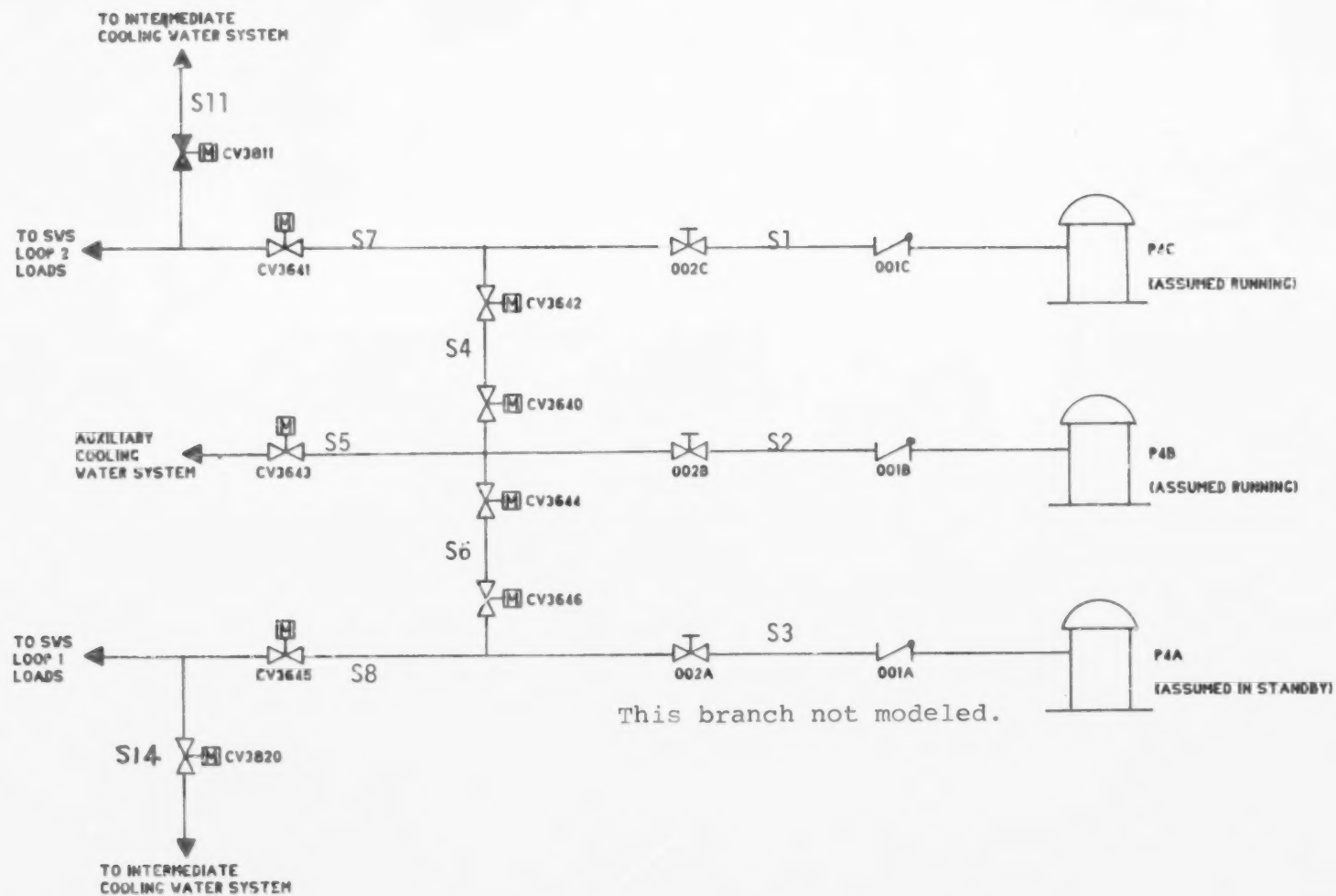


Figure A.19 Pipe Segments for Service Water System Pump Loops.

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There are three SW pumps. During normal operation, two of them are in use with the third pump in standby. All of the crossover valves in the common-pump-discharge header are open, but they close upon ESAS actuation. No valve realignment occurs automatically unless there is an ESAS signal. The ESAS will only send an actuation signal to the two SW pumps that were already running in the normal mode. Success for the SWS is that all components requiring cooling in a specific accident sequence receive sufficient SWS flow for that cooling. The success criteria of the SWS in this analysis are two-fold: that with an ESAS signal present, and that without. With an ESAS condition no credit is given for one loop backing up the other. That is, the loops are designed to isolate on an ESAS signal, and if they do not, the operator is trained to isolate them. In addition, any diversion from a loop subsequent to the ESAS signal, is assumed to fail that loop. For the case without an ESAS condition (or prior to one being initiated), credit is given for one loop backing up the other, and diversions to normal plant loads do not fail the SWS because the pre-ESAS loads are not as large.

The success criteria defined above are the major analysis assumptions for this system. The criteria are conservative in that any diversion in the post-ESAS alignment is a failure. These criteria are not those given by the plant. The criteria given by the plant were numerous because they were dependent on specific actions of sets of components. From the success criteria provided by the plant, the two criteria used in the analysis were identified as containing all the possibilities in an exact or conservative fashion. That is, the criteria used are the more general representation of those of the plant. Furthermore, the use of the developed success criteria aided the sequence analysis. If the criteria of the plant had been used, not only would the specific criterion be dependent on the sequence, but it would also be dependent on the individual, minimal cut sets within a sequence.

11.0 EMERGENCY POWER SYSTEM (EPS)

Emergency AC Electrical System - The emergency AC electrical (EAC) system provides electric power to the ESF equipment of the mitigating systems. The EACS is composed of two trains (see Figure A.20), each comprising a diesel generator, 4160 V switchgear, 480 V load centers and motor control centers, 120 V instrumentation panels, and associated transformers and circuit breakers. The normal power supply to the two trains is offsite power with each train having its own offsite connection at the 4160 V switchgear (A3 or A4).

There are three independent undervoltage sensing circuits per train. Two are connected to A3 and A4, and one to the 480 V load centers B5 and B6. Upon sensing an undervoltage condition, any of them will transmit a signal to open the 309 breaker for



A3 (409 for A4), start the diesel in the train, and close breaker 308 (408 for A4).

Each diesel generator is equipped with a permanent magnet, and if the air start can crank a diesel to 300 rpm, it will start without field flashing from the DC power system. The diesels have a maximum rated starting time of 15 seconds from the admission of the starting air. DC power is required, however, to open the solenoid valves in the lines connecting the air start tanks to the diesel. They cannot currently be manually opened. There are two independent compressor tanks for each diesel. Subsequent to the start, DC power is required for diesel generator and circuit breaker control power. SWS cooling is also required (see Figure A.21).

As can be seen in Figure A.20, A3 and A4 (and B5 and B6) are capable of being cross-tied. The cross-tie breaker sets are designed so that only 3 of the 4 can be closed at any time, and an undervoltage signal severs the tie, if it exists. Normal practice at ANO-1 is not to cross-tie the two trains.

Shift relief sheets at ANO-1 state that B56 is to be normally tied to B6. This can result in having one DC train, and the complementing diesel air start valves being dependent on the other train of the AC and DC systems. For conservatism, this dependent alignment was chosen for the analysis so that its effect could be ascertained.

At ANO-1 there is no load sequencing system per se. Rather, ESAS sends out start signals to all components simultaneously, and the control circuitry of each major load component contains a time delay coil. Conservatively, for this analysis, it is assumed that a failure in these coils will cause the failure of the diesel which ultimately powers that coil. In addition to the ESAS time delay coils, each component has similar, parallel circuitry for an undervoltage signal.

It is assumed that the A3 and A4 switchgear rooms will eventually require cooling. This cooling is modeled as a support system fault of the long running equipment (e.g., pumps, fans) and not for the initially acting components (valves). The modeling of the service water cooling is the same. The reason is that these are failures of the electrical system after time has elapsed (five minutes for lack of service water, several hours for room cooling) which is not a failure mechanism for initially acting components.

The success criteria for this support system are that each bus is powered when ESF equipment requires it.

DC Power System - The 125 volt DC system provides continuous power for control, instrumentation, reactor protection and engineered safeguards actuation systems, and emergency

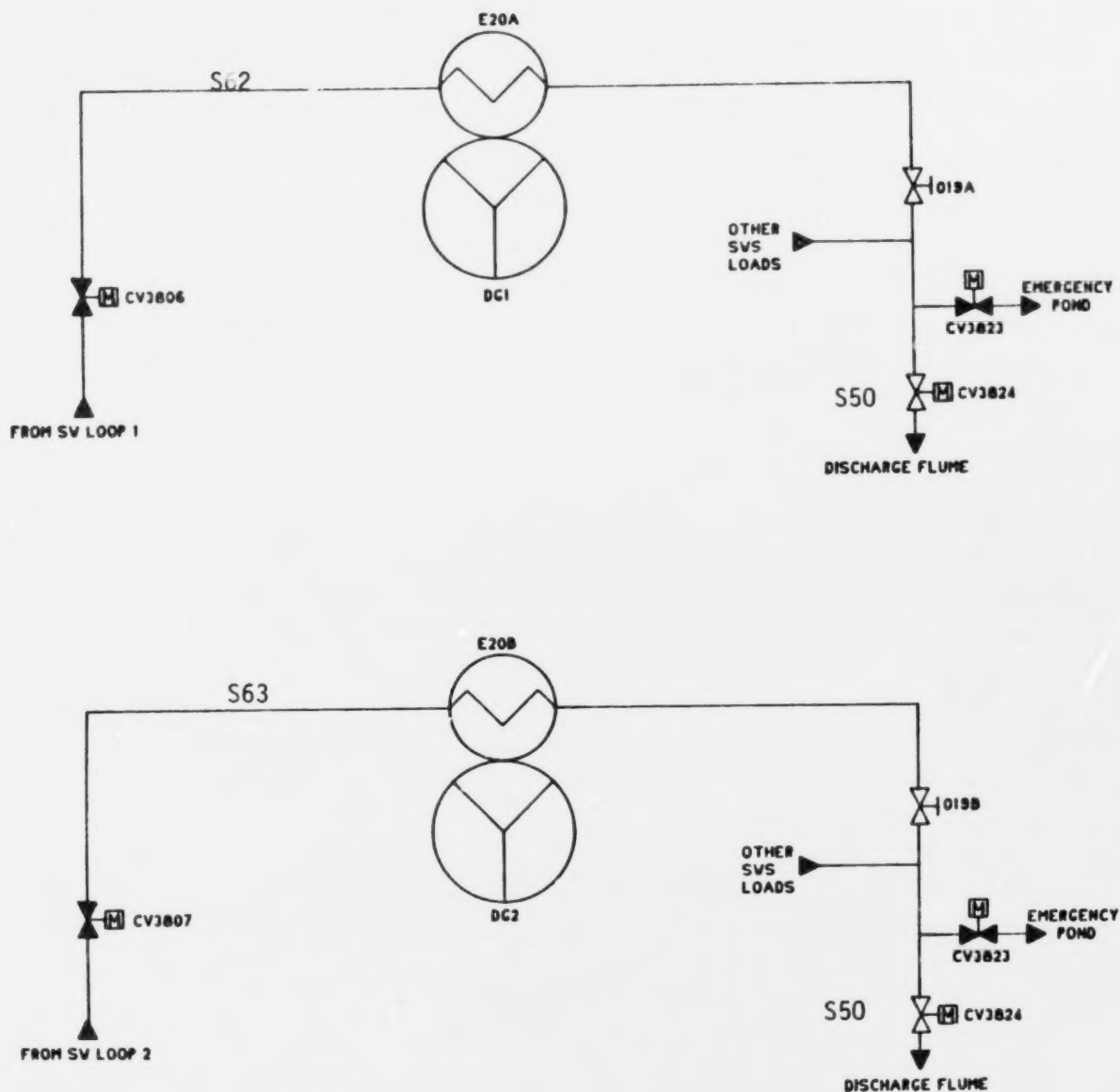


Figure A.21 Pipe Segments for Service Water Cooling of Diesel Generators

safeguard actuation control (e.g., pumps) systems. In addition, it powers the control valves in the emergency feedwater system and provides control power for the diesel generators in the emergency AC electrical system.

The DC system is composed of two separate trains, each comprising a 125 volt battery, bus, and control panels (see Figure A.22). In addition to the batteries, the DC system is also supplied from the emergency AC electrical system via three battery chargers, with two in service at any given time. The charger alignment is rotated monthly.

The busses cannot be cross tied. If maintenance is required on one train, its respective distribution panel can be manually aligned to receive power from the main bus of the other train. The inverters each have three sources of power: the respective DC bus, rectified and transformed 480 V AC from the emergency AC system, and transformer 480 V AC from the emergency AC system which does not actually pass through the inverter itself (see Figure A.23).

Each battery is designed to carry the continuous DC and vital AC loads for a minimum of eight hours following a station blackout. This time could be extended with the manual shedding of loads per the appropriate emergency procedure. Of crucial importance to system performance is the reliability of the batteries. Discharge tests are performed at the refueling shutdown every 18 months. The rest of the time the batteries are floating on the chargers so that it cannot be precisely determined what the capability of the batteries is except at discharge. (Exceptions to this occur monthly when the chargers are rotated. Momentarily the DC system is powered by the batteries alone so that severe degradation would be detected then.) Other inspections include daily checks of the pilot cells, quarterly checks of all the cells and terminals, and an annual check of the cells as well as terminal cleaning. As a consequence of the IREP study, the quarterly and annual tests are now staggered so as to lessen the human common mode inspection error possibility.

It is currently possible to have both in service battery chargers receiving AC power from the same bus. B61 of the AC system is always fed from bus B6, and shift checklists stipulate that swing bus B56 should also be unless diesel generator 2 is undergoing maintenance (see Figure A.20). This then results (see Figure A.22) in both battery chargers receiving power from B6.

It is assumed that the battery rooms will eventually require cooling. This cooling is modeled the same as that for the switchgear rooms and the diesel generator service water cooling, which is discussed below.

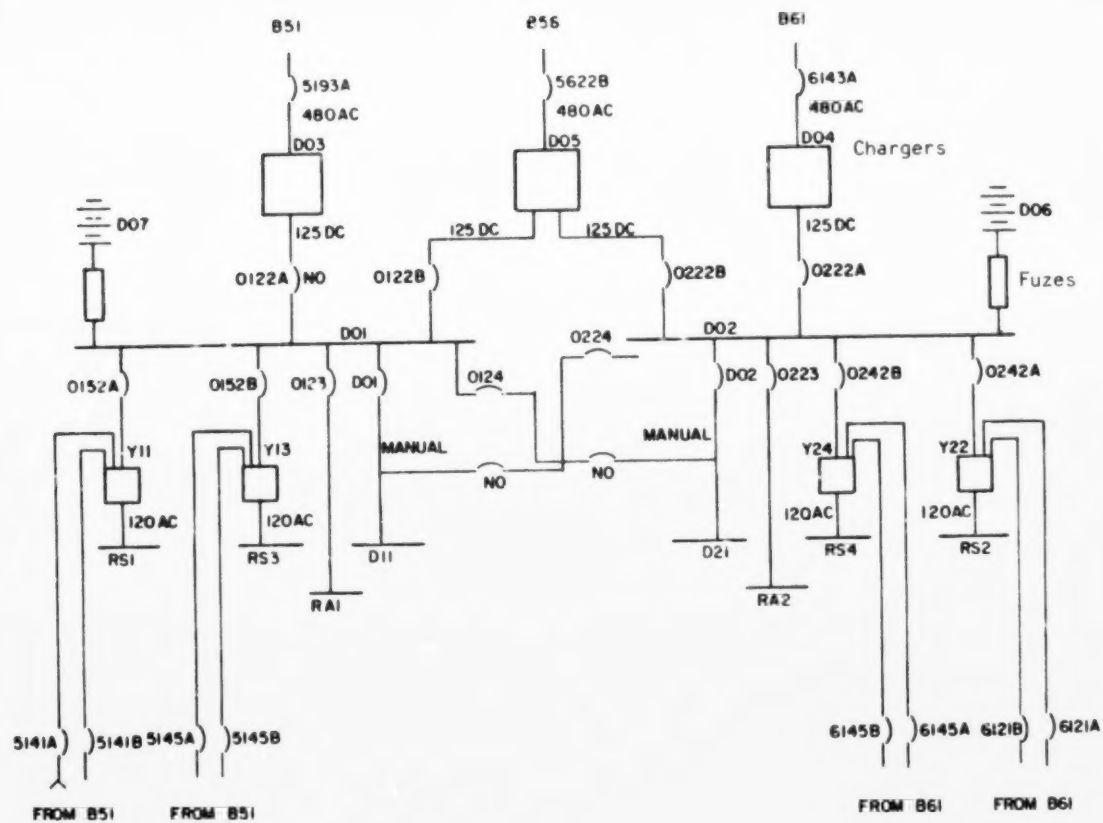


Figure A.22 DC Power System.

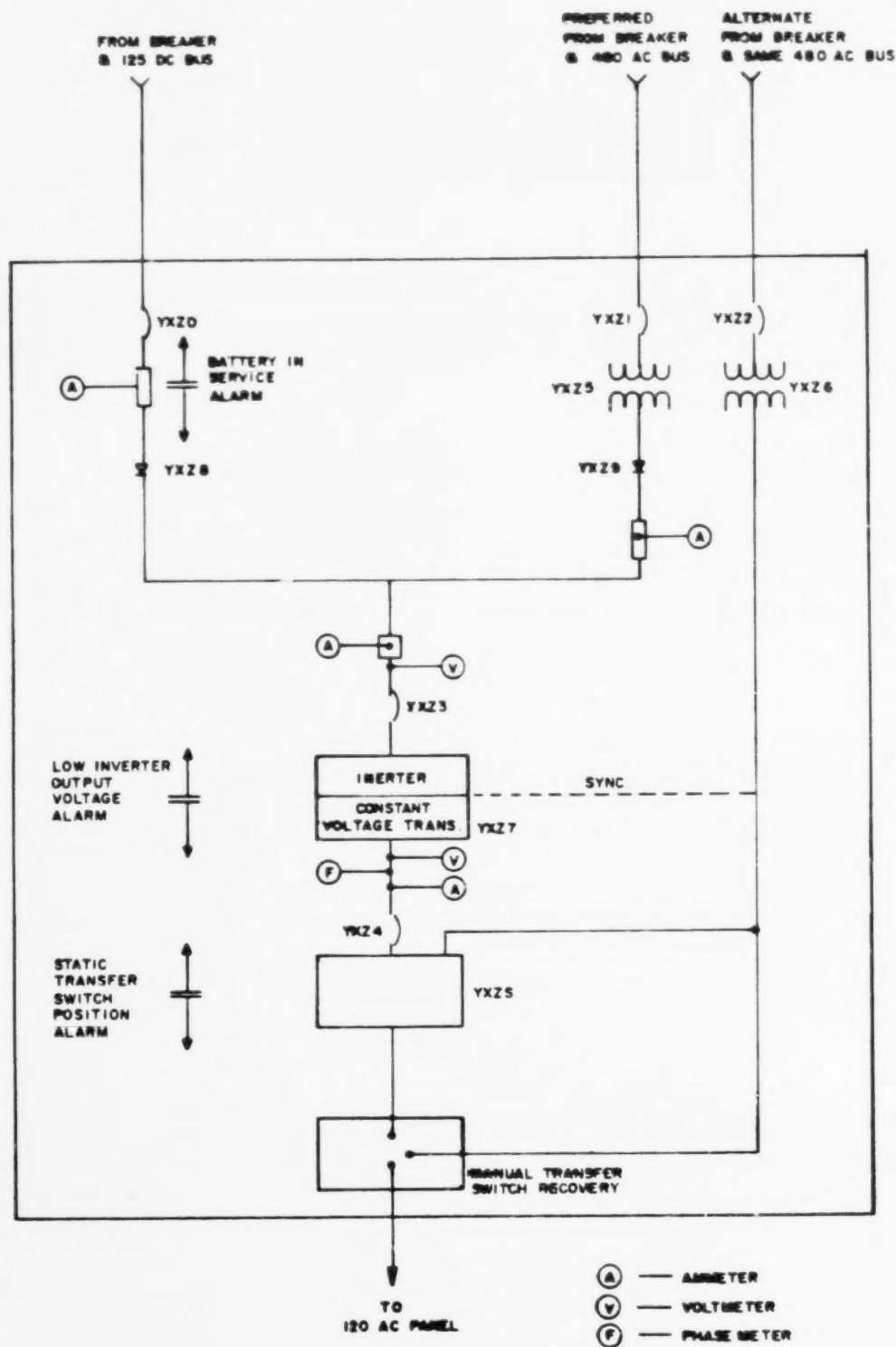


Figure A.23 DC Power System Inverter, with Components Identified as used in the Analysis.

The success criterion for the DC power system is that all components requiring either DC motive or control power receive it. An important assumption in this analysis is that the battery chargers can supply DC power without the batteries, i.e., they are an independent DC power source.

Battery and Switchgear Emergency Cooling System - The purpose of the battery and switchgear emergency cooling system (referred to as ECS) is to provide sufficient cooling to assure that electrical units which must operate during emergency conditions will not fail due to excessive heat. The rooms with electrical equipment in them, which could fail due to overheating, are the two battery rooms and the two switchgear rooms. This is based on an APL analysis which showed that after one hour without cooling the room temperature exceeded that of the duty ambient reading of the batteries. The cooling requirements for the switchgear rooms were not studied in the APL analysis, but for the sake of completeness, the failure of such cooling was assumed in the IREP study to fail the equipment in the switchgear rooms.

The ECS consists of two independent identically configured chilled water trains which provide cooling to the rooms noted above, and two refrigerated air units which provide additional cooling capacity for the north and south battery rooms only. Each chilled water train comprises a chill water unit (CWU), three ventilation unit coolers (VUCs) and associated plumbing. A simplified schematic of the chilled water system is shown in Figures A.24 and A.25. Additional cooling for the north and south battery rooms is provided by self-contained air cooled refrigeration units which are independent of the chilled water system. A single refrigeration system is provided for each battery room. Each system consists of a condenser unit, an evaporator/blower unit and a thermostat. Each unit has sufficient capacity to accommodate 100% of the heat load in its associated battery room.

As a result of the IREP study, the test procedures at ANO-1 were changed to actually test the thermostats under a heat load. Until that time they had never been tested.

For the chill water units the odd ECS train cools odd components of the emergency AC and DC systems as well as being powered by them. Likewise, the even train cools and is powered by the even train. The refrigeration units, however, are both powered by the swing bus of the emergency AC system, which is normally aligned to be fed from B6. Thus, one of the cooling units for the odd train battery is normally powered by an even train component.

The success criteria of the ECS is to provide cooling to each of the four rooms mentioned above. For each of the switchgear rooms this means success of the associated chill water unit.

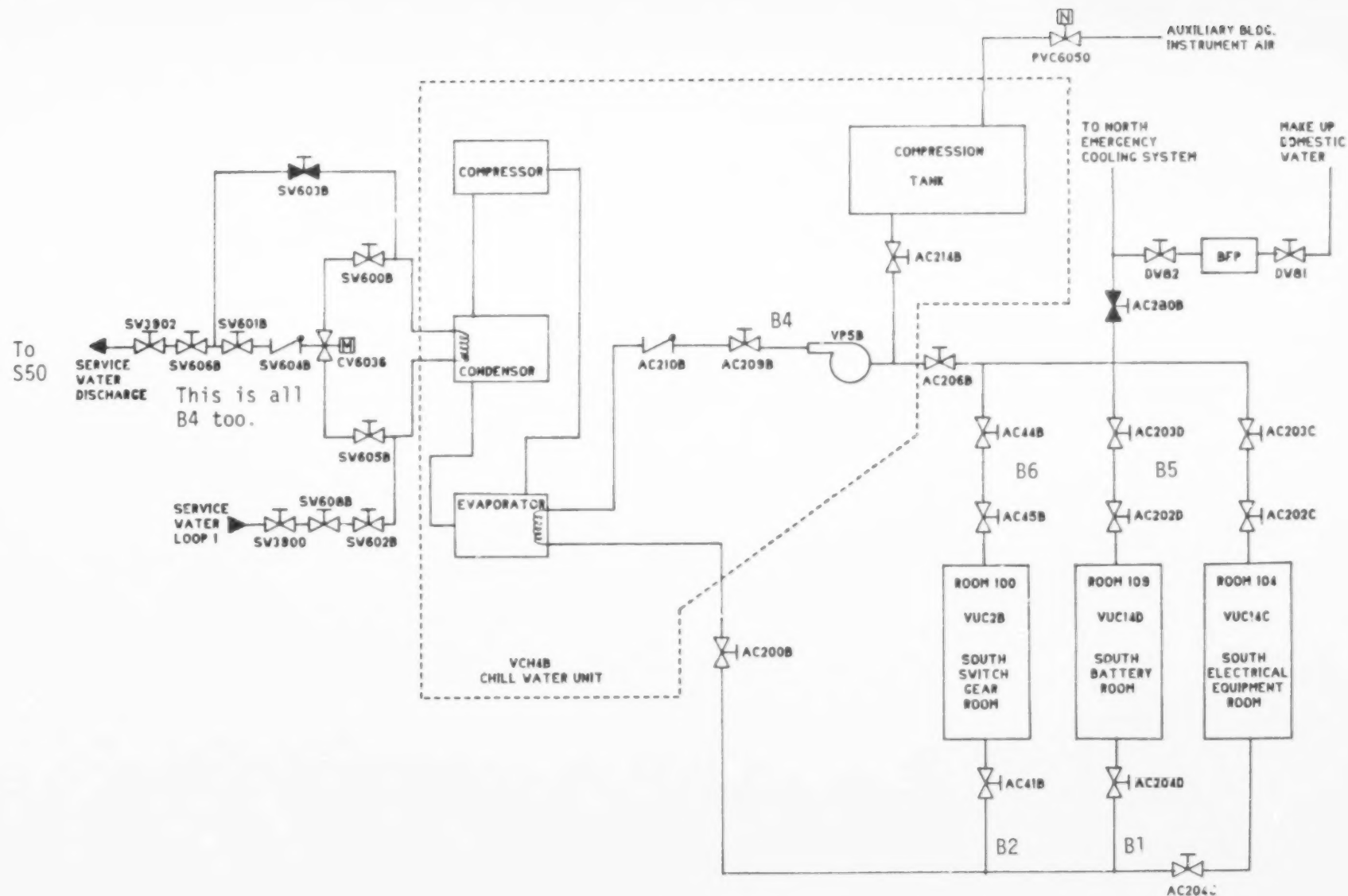


Figure A.24 Pipe Segements for the Emergency Cooling System: South Rooms

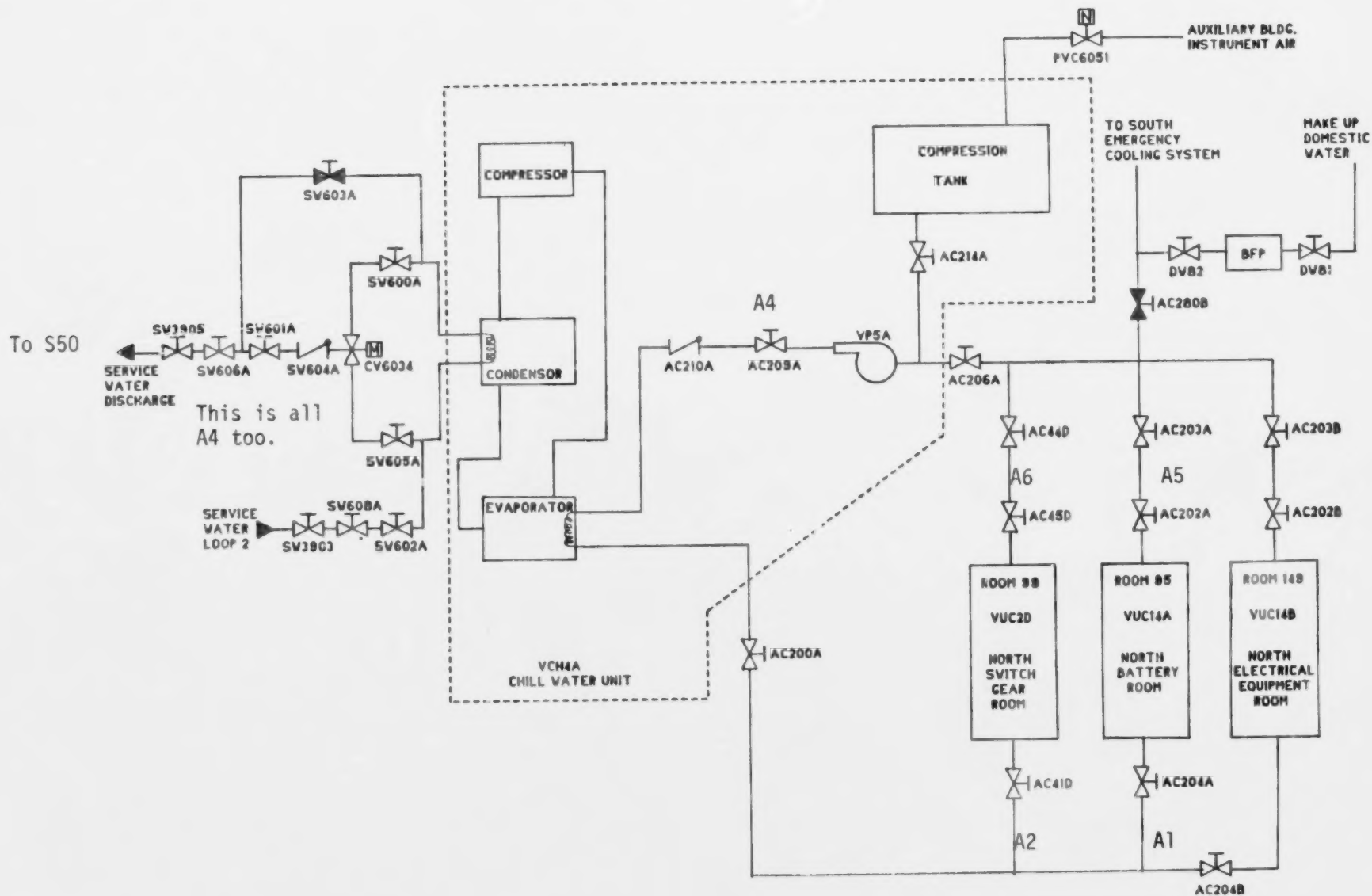


Figure A.25 Pipe Segments for the Emergency Cooling System: North Rooms

For each of the battery rooms these criteria mean that either the associated chill water unit or the refrigeration unit must succeed.

Recent discussions with plant personnel has disclosed that the real purpose of the switchgear and battery room cooling is to extend component life and temperatures given failure of cooling would probably not result in failure of components modeled in TAP A-45. Furthermore, a simple recovery action would be to open the door and possibly set up portable fans to circulate the air. The end result is that generous recovery factors were applied to failures of the battery and switchgear emergency cooling system component in the recovery analysis.

APPENDIX B
INTERNAL ANALYSIS FOR THE
NO-1 NUCLEAR POWER PLANT

by

Wallis R. Cramond
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1.0 INTRODUCTION

The internal analysis involves a number of sequential steps which are described in this appendix. Each step and the section where it is discussed in the appendices or the main report is shown below.

1. Establish System Success Criteria (Appendix B, Section 2.0)
2. Construct Event Trees (Appendix B, Section 2.0)
3. Construct Fault Trees (Appendix A)
4. Establish Data Base (Appendix B, Sections 3.1 and 3.2)
5. Visit Plant for System and Special Emergency Information (Appendix B, Section 5.0)
6. Delineate Accident Sequences (Appendix B, Section 5.1)
7. Perform Core Melt Analysis Using the SETS Computer Program (Appendix B, Section 5.2)
8. Determine Core Melt Dominant Accident Sequences (Appendix B, Section 5.2)
9. Perform Recovery Analysis on Core Melt Dominant Accident Sequences (Appendix B, Section 5.2 and Section 4.0 of the Main Report)
10. Identify Individual Vulnerabilities from Dominant Accident Sequence Cut Sets (Section 4.0 of Main Report and Appendix B, Section 5.2)
11. Propose Modifications for the Most Significant Vulnerabilities (Section 4.0 of Main Report)
12. Integrate Internal and Special Emergency Vulnerabilities (Section 4.0 of Main Report)
13. Determine Potential Alternatives (Section 4.0 of Main Report)
14. Communicate Alternatives to Architectural Engineer (Section 4.0 of Main Report)
15. Visit Plant to Verify Modification Design (Section 4.0 of Main Report)
16. Perform SETS Computer Program Analysis for Combined Core Melt Dominant Accident Sequence and Containment System Sequences (Appendix B, Section 5.5)

17. Reapply Recovery Analysis to Include the Containment Systems (Appendix B, Section 5.5 and Section 4.0 of the Main Report)
18. Assign Dominant Accident Sequences Including Containment System and Containment Failure Mode Probabilities (Appendix B, Section 5.5)
19. Reevaluate Core Melt Probabilities for Each Alternative (Appendix B, Section 6.1)
20. Reevaluate Combined Core Melt and Containment Failure Probabilities and Release Category Assignments for Each Alternative (Appendix B, Section 6.2)

Sections 5.0 and 6.0 of Appendix B is the basic description of the internal analysis while Sections 2.0, 3.0, and 4.0 provide the necessary supporting information.

2.0 EVENT TREE AND SYSTEM SUCCESS CRITERIA

This section of Appendix B describes the ANO-1 event trees and the associated system success criteria. These event trees are related in the following ways. Those branches of the transient event trees that represent transient induced LOCAs, i.e., stuck open pressurizer relief valves, connect to the appropriate branch of the LOCA event tree according to the event successes and failures preceding the induced LOCA. Also, since core melt is not affected by containment systems, the containment event tree can be separate and simply attached to the end of each core melt accident sequence for either the transient or LOCA event trees. These two relationships greatly simplify the event tree presentation and will be discussed further in Section 5.1 as Accident Sequence Delineation and in Section 5.5 on Containment Systems Integration.

2.1 Transient Event Trees

There were some transients which were not in the defined scope of work for this program, e.g., anticipated transients without scram (ATWS) and pressurized thermal shock (PTS). Those transients which are considered can be represented by three fundamental event trees as defined below:

- T₁ - Loss of Offsite Power (LOP) Transient
- T₂ - Loss of Feedwater (LOF) Transient (offsite power is assumed to be available)
- T₃ - All Other Transients (AOT) where the power conversion system is initially available (e.g., Turbine Trip)

Two support system initiating events that will use the same structure as the T₂ event tree are:

T₄ - Loss of an AC Bus which is assumed to trip the unit

T₅ - Loss of a DC Bus which is assumed to trip the unit

There are several features that are common to all the transient event trees.

1. The "feed and bleed" (F&B) mode of primary system cooling is a viable option at ANO-1.
2. It is assumed that no containment overpressure protection is required at ANO-1 for the F&B mode.
3. The high pressure injection system (HPIS) (with backup from the service water system) provides cooling for the reactor coolant pump (RCP) seals, but this is not explicitly included in the transient event trees. It is assumed that the RCP seals will not require cooling. This issue appears to be unresolved at the present time. Also, it is assumed that the HPIS will not be required for coolant makeup while cooling down to stable hot shutdown given a transient with successful secondary cooling.
4. The reactor protection system (RPS) is assumed to succeed in each transient event tree.

2.1.1 Transient System Success Criteria

Each heading in the transient event tree relates to a system or function with an associated success criteria. In a few cases the system is not fault tree modeled, but it is always important to understand what equipment in the system is required for success. Although the support systems are not shown explicitly in the event trees, their success criteria are also given below.

<u>System</u>	<u>Success Criteria</u>
Power Conversion System (PCS) - M	1 of 2 trains (no fault tree model)
Emergency Feedwater Systems (EFWS) - L	1 of 2 trains where one is a motor driven pump (MDP), the other a turbine driven pump (TDP)
Pressurizer Safety Relief Valves fail to open on demand (SRVO) - P	1 of 2 safety relief valves given the ERV is blocked

Pressurizer Safety
Relief Valve fails to
reclose given one has
opened (SRVR) - Q

It is assumed one safety relief
valve opens and thus must
reclose when the pressure
decreases

High Pressure Injection
System (HPIS) - E

1 of 3 safety injection pump
trains. This is part of the
F&B fault tree (E) where
operator initiation of the
system is required.

Service Water System (SWS) 1 of 3 SWS pumps

2.1.2 T₁ - Loss of Offsite Power LOP Transient Event Tree

The T₁ event tree is given in Figure B.1 with the accident sequences and the core status; OK indicating success or no core melt, late core melt (LCM) or early core melt (ECM), and LOCA for those sequences which transfer to the LOCA event tree. Only the core melt or transient induced LOCA accident sequences are numbered.

Success of the RPS is assumed. Failure of the PCS results from loss of offsite power. Thus, the significant part of the event tree starts with the success/failure of the EFWS. Given success of the EFWS, the SRVs are not required to open but may inadvertently open and then be required to reclose. Failure to reclose (SRVR) results in a transient induced LOCA where L is successful (Sequence 1). Successful reclosing is an OK sequence.

If the EFWS fails the safety relief valves will be challenged and failure to open (SRVO) causes a rapid RCS overpressure and an assured reactor vessel rupture without further mitigation options resulting in core melt (Sequence 5). If, however, the safety relief valves open, the pressure is relieved and the next branch is success or failure of the safety relief valves closing. Failure of the SRVs to reclose results in a transient induced LOCA where L has failed (Sequence 4). For successful SRV reclosure, it is then necessary to consider another method to remove decay heat since the EFWS has failed previously in this scenario.

Feed and bleed (F&B) is the next choice with failure of HPIS leading to core melt (Sequence 3). Success of the F&B mode does not require subsequent long term cooling or makeup from the HPRS.

2.1.3 T₂ - Loss of Feedwater Transient Event Tree

The T₂ event tree structure is exactly the same as the T₁ event tree. The only difference is the frequency of the initiating event and various system failures because of the

T ₁	K	M	L	P	Q	E
LOP	RPS	PCS	EFWS	SRVO	SRVR	HPIS

CORE STATUS ACCIDENT SEQUENCE SEQUENCE NUMBER

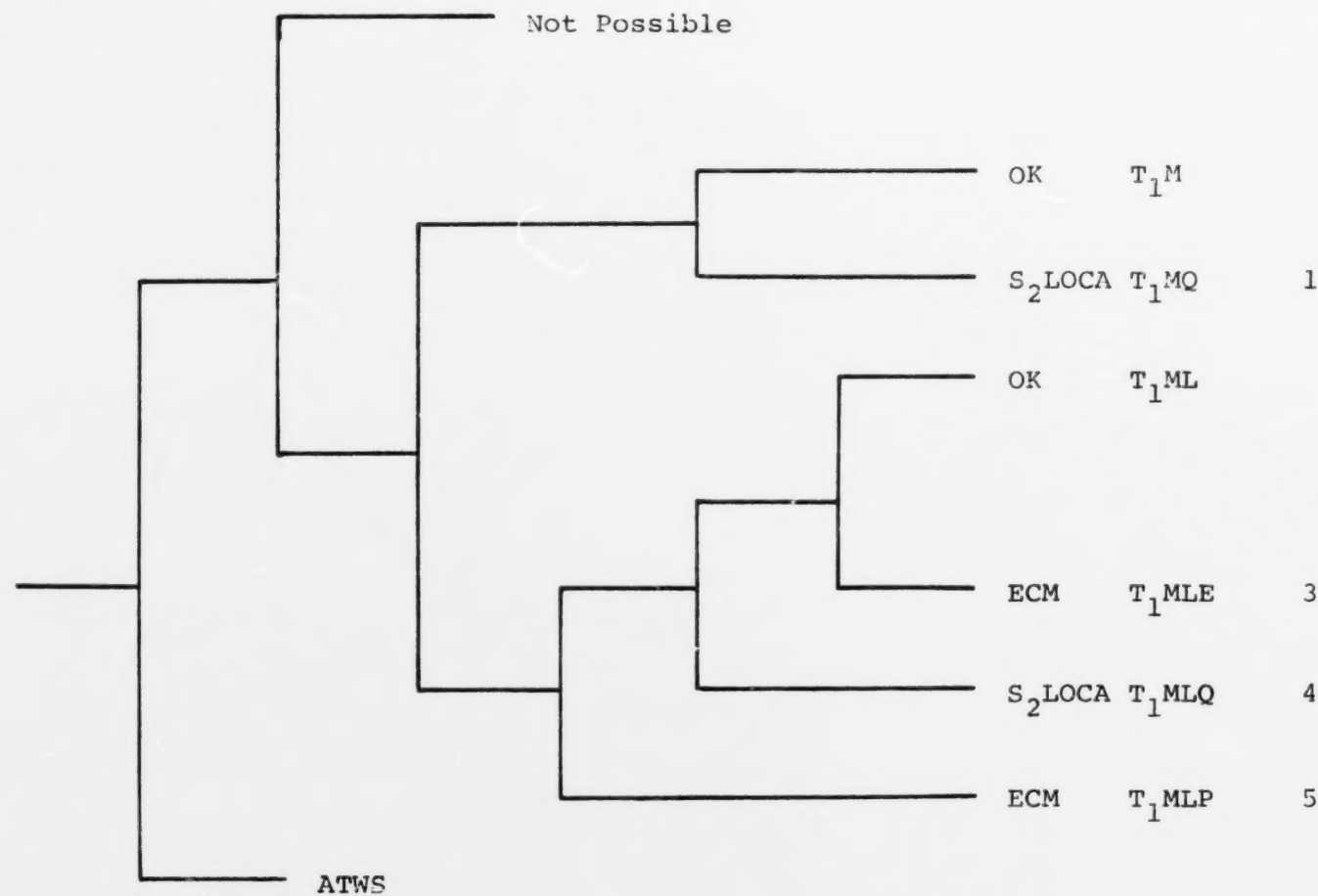


Figure B.1 Loss of Offsite Power Transient (T₁) Event Tree

initial conditions. The discussion of sequences in Section 2.1.2 applies to the T_2 event tree. The T_2 event tree is given in Figure B.2 for completeness. This event tree also applies to T_4 and T_5 .

2.1.4 T_4 - Loss of AC Bus and T_5 - Loss of DC Bus

These bus failure initiating events are special events which are assumed to cause a reactor trip and also leave the unit in a degraded state to respond to the required shutdown. The A3 4160 volt AC bus was selected for T_4 to represent the loss of an AC bus. The D01 DC bus was selected for the DC bus example. In both cases interruption of feedwater is assumed.

2.1.5 T_3 - All Other Transients

This transient category is distinguished from T_1 and T_2 by the power conversion system being initially available. The T_3 transient addresses a large number of transients where the initial conditions are essentially the same, i.e., offsite power is available and the power conversion system is initially available. The event tree is given in Figure B.3.

The discussion found in Section 2.1.2 for the T_1 transient is applicable except that the system failure frequencies will be different and there is one additional branch. This branch starts with PCS(M) success and thus the AFWS is not called upon. If one of the SRVs inadvertently opens, failure to reclose results in a transient induced LOCA (Sequence 2) which transfers to the LOCA event tree where a choice on L is required. Even though the PCS was initially available it is interrupted for this transient induced LOCA. If the SRVs reclose, this becomes an OK sequence.

2.2 Small LOCA Event Tree

The small LOCA event tree for ANO-1 is shown in Figure B.4. It has two important features; a branch where credit is given for feed and bleed, and a branch where credit is given for secondary blowdown. These will be discussed further later in this section.

The defined scope of the program excludes large LOCAs (A), medium LOCAs (S_1), reactor vessel rupture LOCAs (R), interfacing system LOCAs (V), and double steam generator tube ruptures. Single steam generator tube ruptures are considered as part of S_2 since that steam generator can be isolated. Small steam line breaks are considered in T_3 . They can either be isolated or will provide the secondary steam relief instead of the atmospheric relief valves or the turbine bypass valves.

T ₂	K	M	L	P	Q	E
LOF	RPS	PCS	EFWS	SRVO	SRVR	HPIS

CORE
STATUS

ACCIDENT
SEQUENCE

SEQUENCE
NUMBER

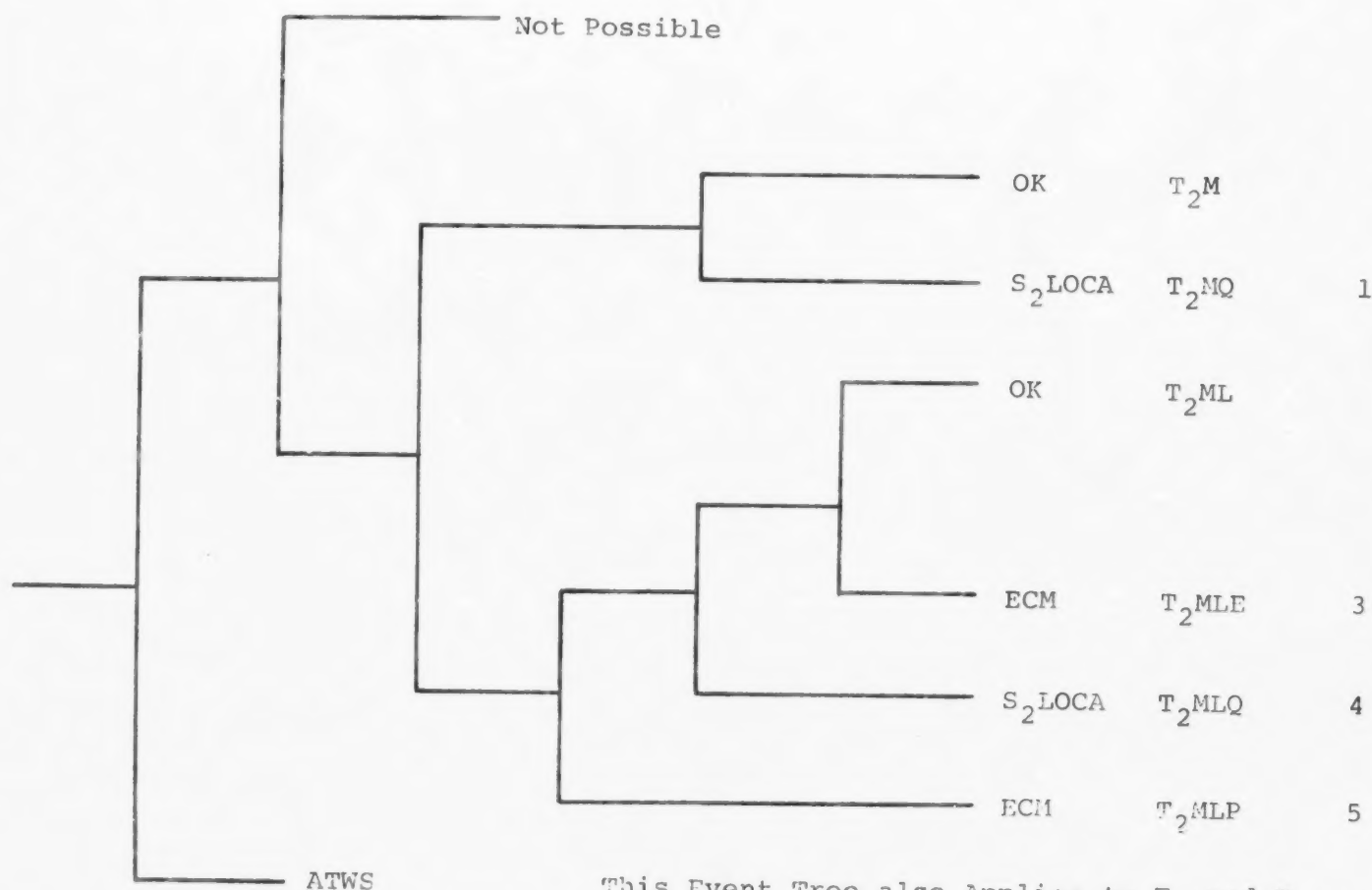


Figure B.2 Loss of Feedwater Transient (T₂) Event Tree

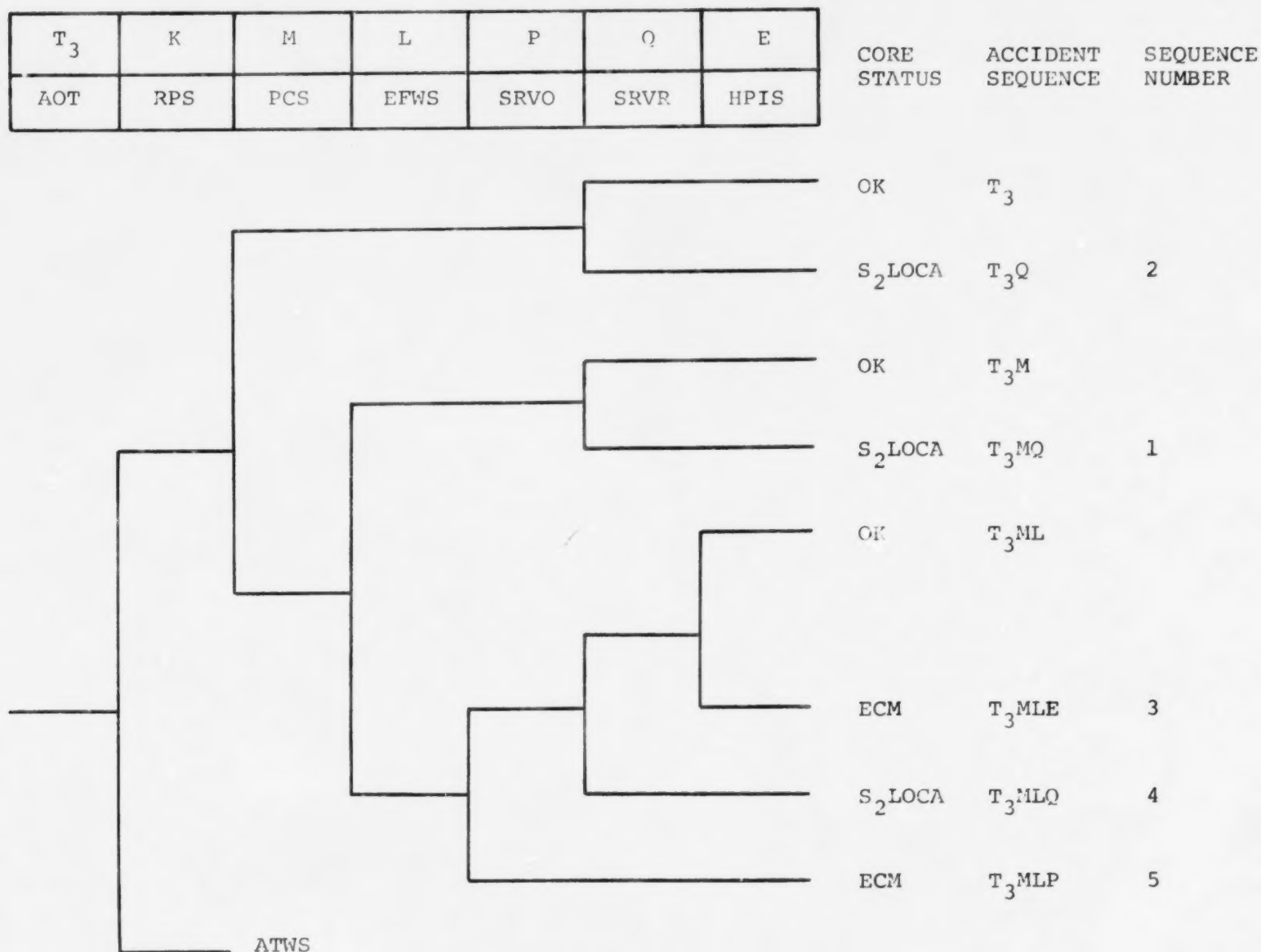


Figure B.3 All Other Transients (T₃) Event Tree

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S ₃	K	M	L	E	P ₁	X	D ₂	H ₁	H' ₂
S ₂	K	M	L	D ₁	P ₁	X	D ₂	H ₁	H' ₂
LOCA	RPS	PCS	EFWS	HPIS	SRV & ERV	SBD	LPIS	HPRS	LPRS

Core Status Accident Sequence Sequence Number

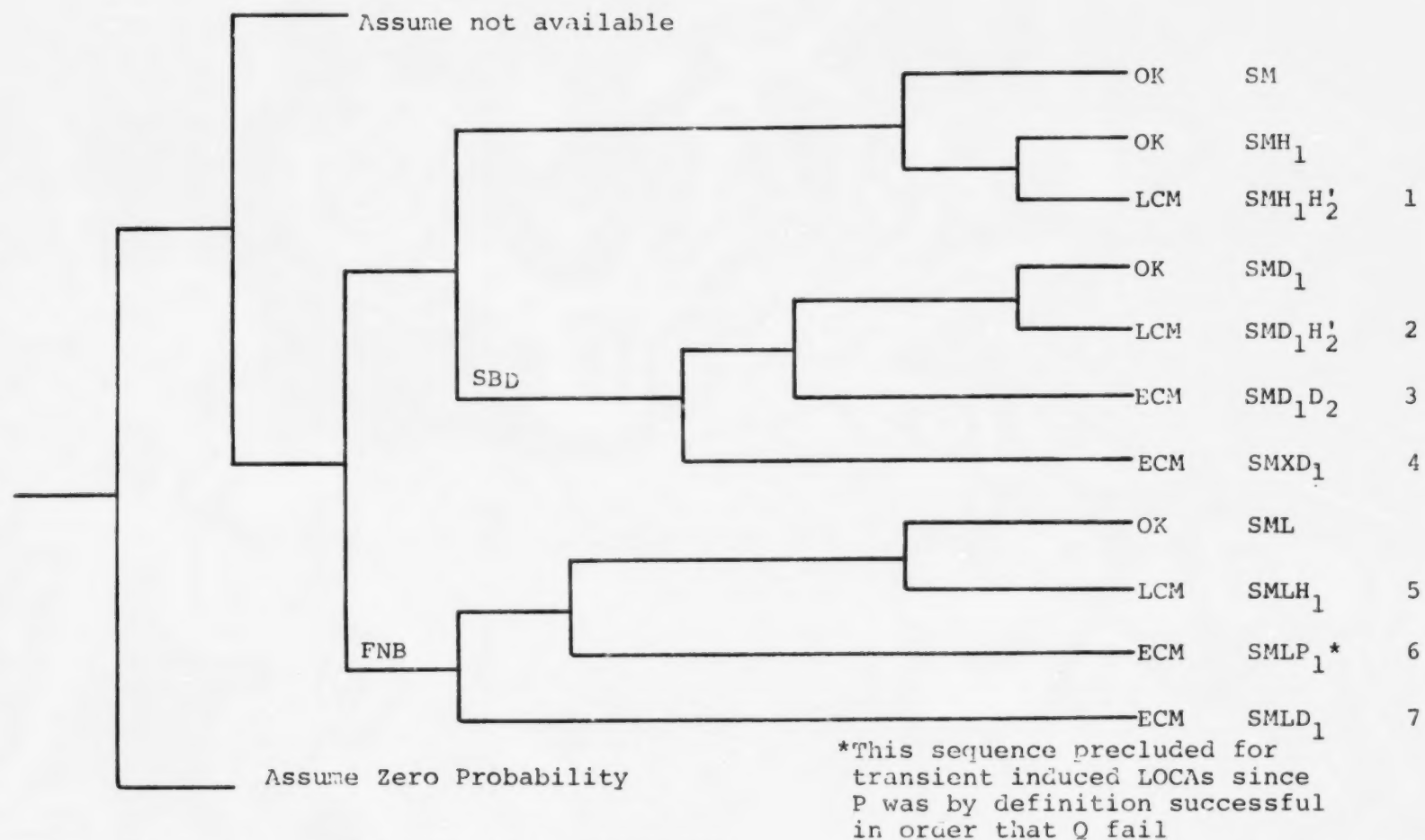


Figure B.4 Small LOCA (S₂) Event Tree

2.2.1 Small LOCA System Success Criteria

Each heading of the LOCA event tree relates to a system or function with an associated success criteria. In a few cases the system does not have a fault tree model but success criteria are given below for all the systems implicitly or explicitly involved in the headings.

<u>System</u>	<u>Success Criteria</u>
Power Conversion System (PCS) - M	1 of 2 trains (no fault tree model)
Emergency Feedwater System (EFWS) - L	1 of 2 trains where one is a motor driven pump (MDP) and the other is a turbine driven pump (TDP)
High Pressure Injection System (HPIS) - D ₁	1 of 3 safety injection pump trains with automatic initiation
HPIS - E	Same as D ₁ except operator initiation
Safety Relief Valves (SRV) - P ₁	1 of 3 SRVs are required for F&B mode of operation
Secondary Blowdown Mode (SBM) - X	Normal pressurizer spray and steam bypass valves or 2 of 2 atmospheric dump valves
Low Pressure Injection System (LPIS) - D ₂	1 of 2 RHR pump trains
High Pressure Recirculation System (HPRS) - H ₁	1 of 3 safety injection pump trains in series with 1 of 2 RHR pump trains and 1 of 2 RHR heat exchangers
Low Pressure Recirculation System (LPRS) - H ₂	1 of 2 RHR pump trains without the RHR heat exchangers
Service Water System (SWS)	1 of 3 SWS pumps

2.2.2 S₂-Small LOCA Event Tree

The core status outcomes in Figure B.4 are: OK indicating success or no core melt, late core melt (LCM), and early core melt (ECM).

The event tree effectively starts with the success/failure of the AFWS since the RPS(K) is assumed to be successful and PCS(M), i.e., main feedwater, will trip from a safety injection signal. Main feedwater is recoverable and will be treated as such later in the analysis. If the EFWS(L) is successful, the next consideration is the HPIS (D_1). HPIS success then requires recirculation to prevent core melt. Given either HPRS (H_1) or LPRS (H_2') success, this leads to an OK core status. Failure of both HPRS (H_1) and LPRS (H_2') leads to a late core melt (Sequence 1). Information from the ANO-1 plant indicates that sufficient water from an S_2 LOCA will be channeled to the sump to assure an adequate recirculation water source. Furthermore, the RHR heat exchangers are not required for H_2 since the EFWS is successful and is adequate to remove all decay by itself. However, the RHR heat exchangers are required for H_1 to keep the HPIS pumps from overheating.

The second major branch of the event tree occurs when the EFWS(L) is successful but the HPIS(D_1) fails. In this case the option that would be selected in the ANO-1 emergency operating procedures is the secondary blowdown mode (SBD) labeled X in the event tree. This requires both pressurizer sprays and secondary steam relief by steam bypass to the condenser or opening 2 of 2 atmospheric dump valves. The objective of this mode is to rapidly depressurize the primary so that the LPIS can be brought into service. Subsequent success of the LPIS and the LPRS later in the accident results in an OK sequence. Failure to depressurize leads to an early core melt (Sequence 4). Successful depressurization but failure of LPIS also leads to an early core melt (Sequence 3). Successful depressurization and LPIS but subsequent failure of the LPRS results in a late core melt (Sequence 2).

The third major branch of the S_2 event tree is where the EFWS(L) fails. Since feed and bleed is a viable option in transient situations where the EFWS fails, it is likewise applicable to the small LOCA. The HPIS gets a safety injection signal due to a S_2 LOCA so an operator initiated HPIS start is only required for a S_3 LOCA, i.e., event E vs event D_1 .

To continue with the feed and bleed branch of the small LOCA event tree, failure of the HPIS(D_1 or E) leads directly to early core melt (Sequence 7). Given success of HPIS, failure to open both SRVs(P_1) also leads to an early core melt (Sequence 6). Finally, even with success of HPIS and P_1 , recirculation phase cooling is required with the RHR heat exchangers since the EFWS has already failed. Thus, failure of H_1 after success of HPIS and P_1 leads to a late core melt (Sequence 5). The LPRS is not considered as a backup to the HPIS since the pressure probably will not decrease sufficiently to permit LPRS operation. Note, however, that the LPRS is needed for success of the HPRS and is modeled that way.

2.3 Containment Systems Event Tree and Success Criteria

Past PRAs have usually assumed that failure of the containment due to failure of the containment systems to protect against overpressure led to sump flashing and subsequent low pressure (RHR) pump cavitation. At ANO-1, and perhaps most other PWRs, it is estimated that at most only a foot or so of sump water would flash before the sump water would cool enough to come into equilibrium with the containment environment. Thus, the containment systems can be separated from the core melt event trees and applied in tandem to each accident sequence. In practice it is only necessary to consider the containment event tree for core melt sequences, and further, only to dominant core melt accident sequences. This tends to simplify the analysis process. The containment system event tree is given in Figure B.5.

2.3.1 Containment Systems Success Criteria

Each heading in the containment systems event tree relates to a system or combination of systems. Each system has a success criteria and a fault tree model. The success criteria are given below.

<u>System</u>	<u>Success Criteria</u>
Containment over pressure protection:	
Reactor Building cooling fans (RBCF) - Y	1 of 4 fan trains
or	
Reactor Building spray injection (RBSI) - C	1 of 2 trains during injection phase
Reactor Building spray recirculation (RBSR) - F	1 of 2 trains during recirculation phase plus sump mixing with 1 of 3 HPRS and 1 of 2 LPRS w/heat exchangers (H ₁)
Post accident radio activity removal:	
Reactor Building spray injection (RBSI) - C	1 of 2 trains during injection phase
Reactor Building spray recirculation (RBSR) - F	1 of 2 trains during recirculation phase

2.3.2 Containment Systems Event Tree

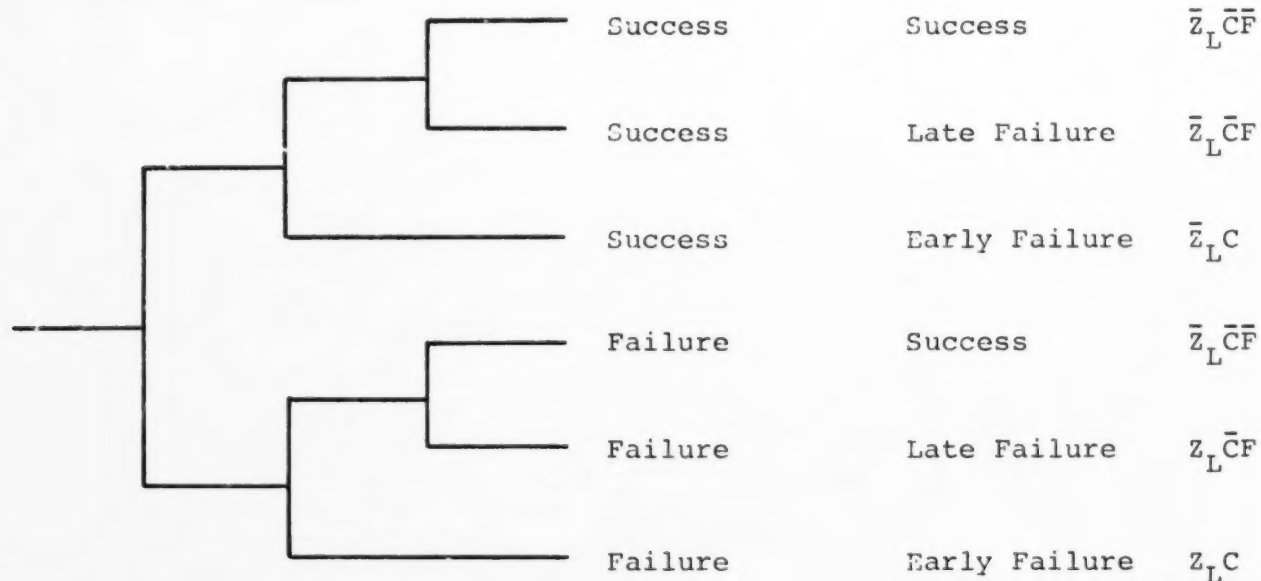
Figures B.5a and B.5b give the simplified containment system event tree (CSET) for ANO-1. This is not to be confused with the traditional containment failure mode event tree which will be discussed later. The purpose of the CSET is to establish

Core Melt Sequence	Z_L	C	F
	COP	RBSI	RBSR

Status of
Containment
Overpressure
Protection (COP)

Status of
Post Accident
Radioactivity
Removal (PARR)

Containment
Systems
Sequence (CSS)



$Z = Y(C+F+H_1)$ all accident sequences except those containing H_2'

$Z = Y(C+F+H_2')$ accident sequences containing H_2'

where $H_1 \equiv 1/3 \text{ HP} + 1/2 \text{ LP} + 1/2 \text{ HTX}$ $H_2' \equiv 1/2 \text{ LP}$

$\therefore$ assume $Z_L = Y(C+F+H_1)$

Figure B.5a Containment Systems Event Tree - LOCA

Core Melt Sequence	Z_T	C
	COP	RBSI

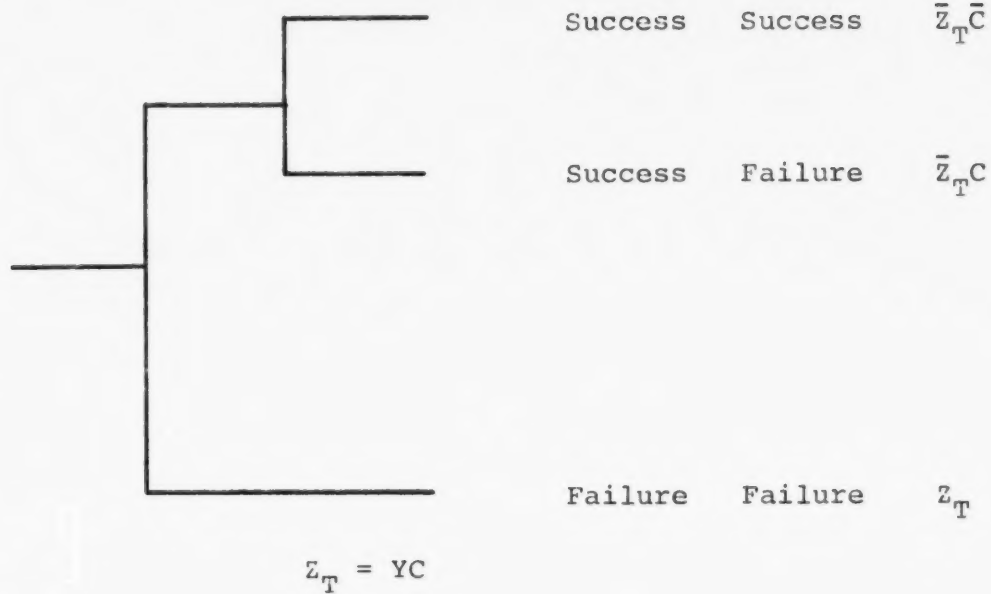


Figure B.5b Containment Systems Event Tree - Transient

the containment state which combined with the core melt accident sequence provide the complete systems accident sequence and are the bases for mapping into release categories.

It is not necessary to trace the containment event tree branch by branch since it is symmetrical and the status columns in Figure B.5 are self-explanatory. The first success/failure branch is on containment overpressure protection (Z). The function Z is defined as

$$Z_L = Y(C+F+H_1) \text{ for LOCAs and } Z_T = YC \text{ for transients}$$

What this says is the containment can be protected from overpressure by

1. 1 operating containment fan unit, or
2. for LOCAs 1 of 2 RBS trains in injection and recirculation plus sump cooling (H_1)
for transients 1 of 2 RBS trains in the injection phase since the recirculation phase should not become necessary.

While the event tree could have been drawn to show each system explicitly, this would provide no additional useful information.

The success as well as failure events are shown in the containment system sequence column. Even the success cases are important since given core melt the containment could still fail and it is the way it fails and the probability of failure that are important to risk to the public.

3.0 DATA BASE

3.1 Initiating Events and Events Not Modeled

Part of every accident sequence involves events that are not quantified using a fault tree model. These events include the initiating event and certain events such as the failure of the PCS or failure of the pressurizer relief valves to open. The frequency or probability of the combination of non-modeled events in a given accident sequence will be called the multiplier probability. An example would be the accident sequence

$$T_3 \text{MQX} \bar{L}D_1$$

where

T_3 is the initiating event,

M, Q, and X are the other non-modeled events, and

$\bar{L}D_1$ is the Boolean accident sequence.

Thus, T_3MQX is the multiplier expression and $p(T_3MQX)$ is the multiplier probability. The non-modeled events are independent so that the individual probabilities can be multiplied together. Continuing the example,

$$p(T_3MQX) = p(T_3) * p(M) * p(Q) * p(X)$$

There are eleven events and one pseudo event (the PORV opening when not demanded) which are described below. In general we will use generic data to permit better comparison between plant types and to avoid any controversy regarding the latest or best plant specific data.

Small LOCA (S_2)

$$f(S_2) = 3.1E-4 \text{ per reactor year}$$

This comes from the IREP Procedures Guide NUREG/CR-2728, page 74, Table 5.3-1. It includes the frequency for 1.2" to 1.66" diameter breaks only.

Small LOCA (S_3)

$$f(S_3) = 2.0E-2 \text{ per reactor year}$$

This comes from the IREP Procedures Guide NUREG/CR-2728, page 74, Table 5.3-1. It covers small LOCA breaks between 0.38" and 1.2" equivalent diameter.

Loss of Offsite Power Transient (T_1)

$$f(T_1) = 8.6E-2 \text{ per reactor year}$$

The frequency given in NUREG/CR-2728, Table 5.3-1, is $3.2E-1$; however, later generic data is available in the EPRI NSAC/80 July 1984 report. In this report the data through 1983 indicate a frequency of 0.088 per site year, and for the last three years alone (1981-1983) a frequency of 0.027 per site year. While the latter number may seem to be the more reasonable frequency to characterize the current situation, a 0.086 frequency as stated in NUREG 1032 will be used for loss of offsite power.

Transient Initiated by a Total Interruption of Main Feed-water (T_2)

$$f(T_2) = 1.0E-0 \text{ per reactor year}$$

This frequency comes from NUREG/CR-2728, Table 5.3-1, and thus will be considered as generic data.

Transient Initiated by a Loss of a Vital AC or DC Bus

$$f(\text{T-AC Bus}) = 6\text{E-}3$$

$$f(\text{T-DC Bus}) = 3\text{E-}3$$

These frequencies are based upon the ANO-1 and ASEP information coupled with our own engineering judgment. This information is summarized below.

AC-BUS INITIATING EVENT

ANO-1 IREP	3.5E-2
Original ASEP	6E-2
Revised ASEP	5E-3
Latest ASEP	5E-3
Other Studies	9E-4 to 6E-2

DC-BUS

ANO-1 IREP	1.8E-2
Original ASEP	6E-2
Revised ASEP	5E-4
Latest ASEP	5E-3
Other Studies	9E-4 to 6E-2

The magnitude of the IREP frequencies seemed high to us so the ASEP frequency was chosen as the basis. Nevertheless, the factor of 2 ratio between the AC and DC Bus frequencies used in IREP seemed reasonable since there are effectively 2 AC Buses for each DC Bus.

All Other Transients Which do not Affect Front Line Systems Significantly (T_3)

$$f(T_3) = 7.1\text{E-}0 \text{ per reactor year}$$

As the name implies, this transient covers everything not included in T_1 , T_2 , T_4 , and T_5 . Such transients occur frequently. The frequency given in NUREG/CR-2728 Table 5.3-1 is 7.1 occurrences per reactor year.

Failure of Power Conversion System and Secondary Steam Relief (M)

$$f(M) = 6.0\text{E-}2 \text{ per demand}$$

This is the failure of the PCS mode of removing decay heat given the PCS is initially available following the transient. This frequency comes from the ASEP data base for B&W plants.

Failure of Depressurization with Secondary Blowdown (X)

$$p(X) = 3.3E-2 \text{ per demand}$$

$$p(X) = 1.0E-0 \text{ per demand given loss of offsite power}$$

Given a LOCA or transient induced LOCA and EFWS success, this mode is used to respond to the situation where HPIS fails. If the primary can be depressurized, the LPIS can then provide injection. This depressurization is accomplished using pressurizer sprays and secondary steam relief or dumping beyond that needed with the EFWS. Although secondary blowdown was not modeled in detail, a simplified fault tree was used to establish the failure probability. The model is shown in Figure B.6. The model is relatively self-explanatory. The Boolean expression using symbols from the fault tree is

$$X = X_1 + (X_2 + X_3) + X_6 [X_7(X_9 + X_{10}) + X_8(X_{11} + X_{12})]$$

Assuming these events are independent and the following event failure probabilities,

$$p(X_1) = 1.5E-2 \text{ typical operator error failure probability } 3E-3 \text{ times } 5 \text{ for stress factor}$$

$$\begin{aligned} p(X_2) &= 1 \text{ for loss of offsite power} \\ &= \text{VOC-LF } (3.6E-5) + \text{VCC } (8E-3) + \text{XOC } (1E-4) \\ &= 8.1E-3 \text{ from sum of three valves} \end{aligned}$$

$$\begin{aligned} p(X_3) &= 1 \text{ for loss of offsite power} \\ &= 1E-2 \text{ estimate of pump unavailability} \end{aligned}$$

$$p(X_6) \quad \text{See analysis}$$

$$p(X_7) = p(X_8) = 1E-2 \text{ human failure to be able to manually jack open these valves}$$

$$p(X_9) = p(X_{11}) = 2E-2 \text{ failure of an electric power train}$$

$$p(X_{10}) = p(X_{12}) = 3E-4 \text{ failure of relief valve to open on demand}$$

The probability of X can be quantified for two cases; loss of offsite power (LOP) for transient induced LOCAs deriving from T₁, and all other initiating events (AOE). Substituting the probabilities from the list above,

$$p(X|LOP) = 1$$

since the unavailability of the pressurizer sprays due to loss of the RCS pumps fails depressurization.

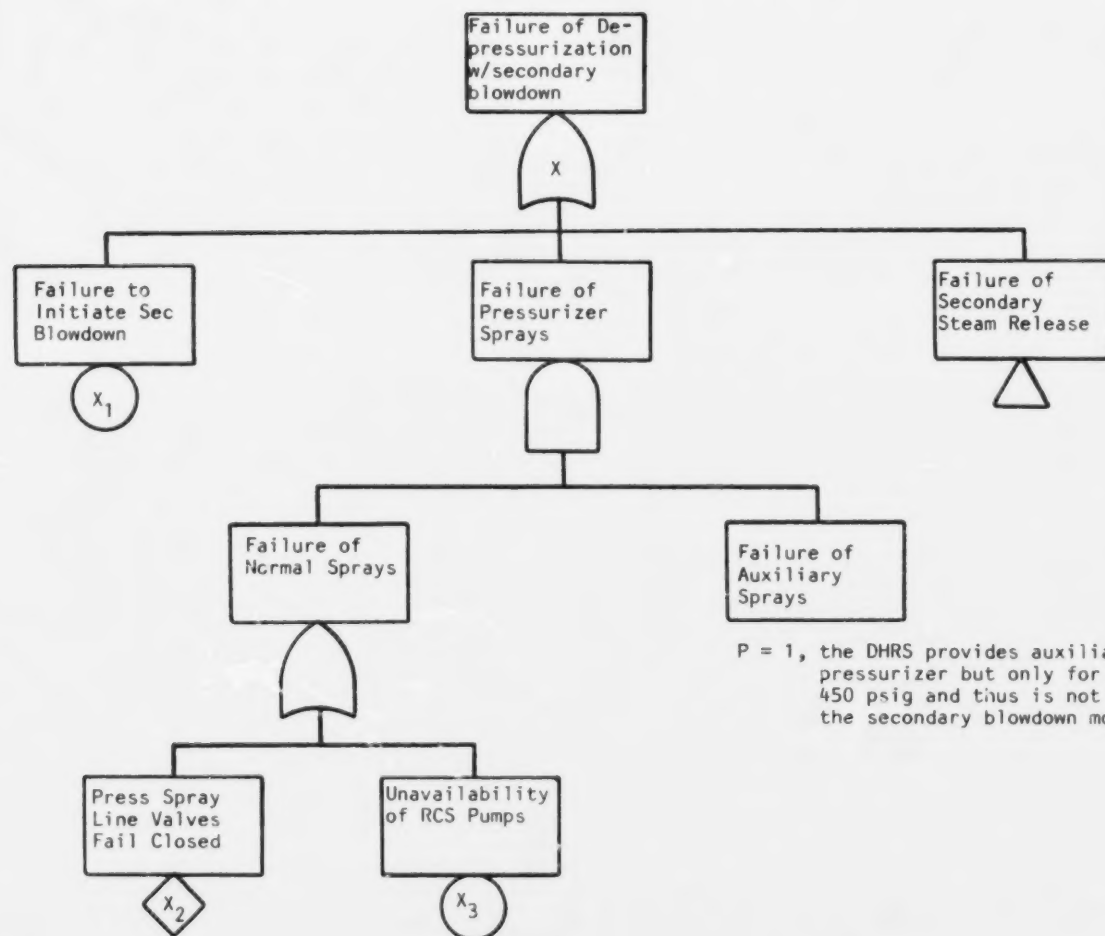


Figure B.6 Simplified Fault Tree for Secondary Blowdown

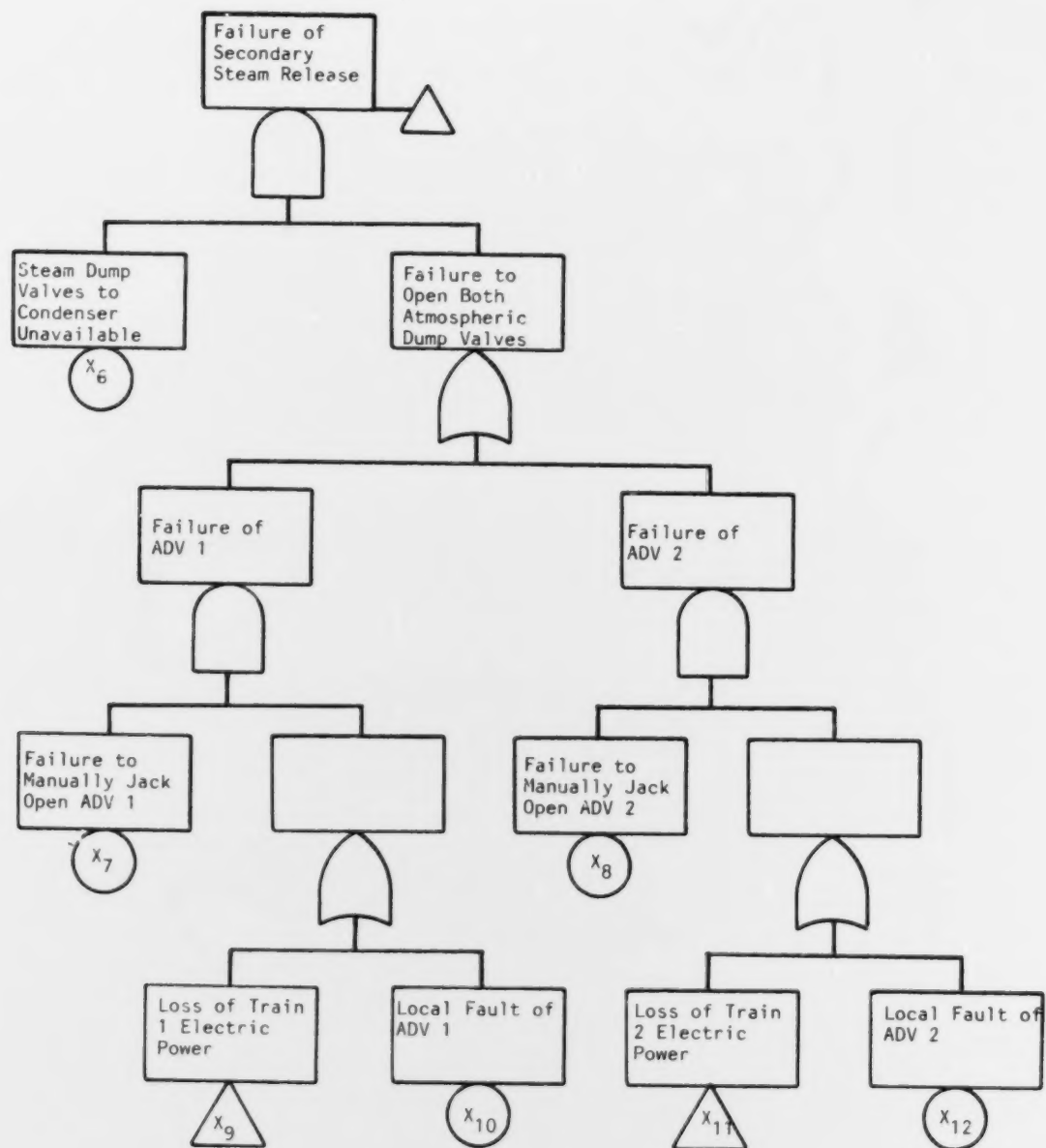


Figure B.6 Simplified Fault Tree for Secondary Blowdown (continued)

$$\begin{aligned}
 p(X|AOE) &= (1.5E-2) + (8.1E-3) + (1E-2) \\
 &+ p(X_6)[(1E-2)(2E-2+3E-4) + (1E-2)(2E-2+3E-4)] \\
 &= (3.3E-2) + p(X_6)(4E-4)
 \end{aligned}$$

Again, if we assume $p(X_6) = 1$,

$$p(X|AOE) = 3.3E-2 \text{ per demand}$$

Frequency of the PORVs Opening When Not Needed - In several branches of the event trees the pressurizer relief valves are not required to open to prevent overpressure but, nevertheless, may inadvertently open. Experience has shown that this occurs about 7% of the time. Thus this factor is used in all those cases to account for the likelihood of the relief valves opening when it is subsequently asked whether the valves closed or not, i.e., event Q.

Failure of the Pressurizer Relief Valves to Close Given That They are Open (Q)

$$p(Q) = 2.0E-2 \text{ per demand}$$

Given that a pressurizer relief valve is open, the likelihood that it would not close or stick open is $1.0E-2$ taken from NUREG/CR-2728, p 127, Table 5.1-1, section 2.7.2. Since there are 2 SRVs the probability is $1.0E-2 + 1.0E-2 = 2.0E-2$.

Failure of the Pressurizer Relief Valves to Open When Demanded (P)

$$p(P) = 1.3E-6 \text{ per demand}$$

There are two cases to be considered given the system has 1 PORV and 2 SRVs. The PORV has a block valve which may or may not be closed during normal operations. Thus Case 1 is where the plant operates with the block valve open and Case 2 is where the plant operates with the block valve closed. The probabilities are taken from NUREG/CR-2728, page 127, Table 5.1-1 as noted in Table B.2 of this report.

Case 1:

$$\begin{aligned}
 p(P) &= p(\text{PORV-LF}) * p(\text{SRV-CM}) \\
 &= (3E-4) * (1.3E-6) = 3.9E-10
 \end{aligned}$$

Case 2:

$$p(P) = p(\text{SRV-CM}) = 1.3E-6$$

Case 2 will be used in the base case analysis.

The preceding discussion is the rationale behind the probabilities used to quantify the multipliers for the accident

sequences evaluated in the ANO-1 internal analysis. Table B.1 shows the multiplier frequency for each non-Boolean Multiplier expression established by the event trees. The complement events ($\bar{X}$ and $\bar{Q}$) were also considered.

3.2 Basic Event Probabilities

The data used for the quantification of the ANO-1 fault trees were collected from a variety of references. Generally, the IREP procedures guide was the source for most hardware failure rates, the Station Blackout Accident Analysis (TAP-A44) was used for most electrical failure rates and the Human Factors Handbook for most failures due to human errors.

The data in Table B.2 was used in the quantification of the ANO-1 fault trees. The table lists the event ID, a brief description of the failure, the failure rate and the reference.

This is generic data and was used to generate the SETS value block in which probabilities are assigned to every fault tree basic event. For example, the event LP11407A-VCC-LF is given an event frequency of $8E-3$ found in Table B.2 under VCC-LF which is the same frequency for any motor operated valve that is normally closed and fails to open on demand, i.e., fails closed. Thus the table represents many more events than are actually shown. Some specific events are given in Table B.2 as necessary.

Some failures are listed as having more than one failure rate. This results from the consideration of the loss of offsite power (LOP) as an initiating event which in turn affects other components in the plant, specifically the service water pumps which are normally running. Since these pumps are kept running during power operations the normal failure rate consists of a failure to keep running. However, given loss of offsite power, these pumps stop and the failure rate becomes the sum of a failure to start and a failure to continue running.

3.3 Failure Probability Rationale

Several event probabilities deserve further explanation than can be conveniently placed in the left hand column of Table B.2.

3.3.1 Battery Failure Rate

ANO-1 conducts a quarterly individual cell voltage and specific gravity test. Such tests, according to NUREG-0666, will detect many of the potential problems that could lead to battery failure, but not all the problems. A partial discharge test would probably be sufficient to detect most of the potential failures not found during the quarterly test. However, the only discharge test conducted at ANO-1 is done on a 1.5 year interval. In addition, when switching between battery chargers

(Text continued on page B-36.)

Table B.1 Frequency of Initiating Events and Event Tree Headings Not Modeled

	S	T ₁	T ₂	T ₃	M	X	P	Frequency SRVs Open When Not Needed	Q	Multi- plier Frequency
S ₂ M	3.1E-4				1.0E-0					3.1E-4
S ₂ M $\bar{X}$	3.1E-4				1.0E-0	.967				3.0E-4
S ₂ MX	3.1E-4				1.0E-0	3.3E-2				1.0E-5
S ₃ M	2.0E-2				1.0E-0					2.0E-2
S ₃ M $\bar{X}$	2.0E-2				1.0E-0	.967				1.9E-2
S ₃ MX	2.0E-2				1.0E-0	3.3E-2				6.6E-4
T ₁ M $\bar{P}\bar{Q}$		8.6E-2			1.0E-0				.98	8.4E-2
T ₁ MP		8.6E-2			1.0E-0		1.3E-6			1.1E-7
T ₁ MQ		8.6E-2			1.0E-0			7E-2	2E-2	1.2E-4
T ₁ MQ $\bar{X}$		8.6E-2			1.0E-0	Zero		7E-2	2E-2	Zero
T ₁ MQX		8.6E-2			1.0E-0	1.0E-0		7E-2	2E-2	1.2E-4
T ₁ M $\bar{P}\bar{Q}$		8.6E-2			1.0E-0				2E-2	1.7E-3
T ₂ M $\bar{P}\bar{Q}$			1.0E-0		1.0E-0				.98	9.8E-1
T ₂ MP			1.0E-0		1.0E-0		1.3E-6			1.3E-6
T ₂ MQ			1.0E-0		1.0E-0			7E-2	2E-2	1.4E-3
T ₂ MQ $\bar{X}$			1.0E-0		1.0E-0	.967		7E-2	2E-2	1.4E-3
T ₂ MQX			1.0E-0		1.0E-0	3.3E-2		7E-2	2E-2	4.6E-5
T ₂ M $\bar{P}\bar{Q}$			1.0E-0		1.0E-0				2E-2	2.0E-2
T ₃ M $\bar{P}\bar{Q}$				7.1E-0	6.0E-2				.98	4.2E-1
T ₃ MP				7.1E-0	6.0E-2		1.3E-6			5.5E-7

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Table B.1 Frequency of Initiating Events and Event Tree Headings Not Modeled (cont.)

	S	T ₁	T ₂	T ₃	M	X	P	Frequency SRVs Open	Q	Multiplier
								When Not Needed		Frequency
T ₃ MQ				7.1E-0	6.0E-2			7E-2	2E-2	6.0E-4
T ₃ MQ $\bar{X}$				7.1E-0	6.0E-2	.967		7E-2	2E-2	5.8E-4
T ₃ MQX				7.1E-0	6.0E-2	3.3E-2		7E-2	2E-2	2.0E-5
T ₃ M $\bar{P}$ Q				7.1E-0	6.0E-2				2E-2	8.5E-3
T ₃ Q				7.1E-0				7E-2	2E-2	9.9E-3
T ₃ Q $\bar{X}$				7.1E-0		.967		7E-2	2E-2	9.6E-3
T ₃ QX				7.1E-0		3.3E-2		7E-2	2E-2	3.3E-4

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Table B.1 Frequency of Initiating Events and Event Tree Headings Not Modeled (cont.)

	T ₄	T ₅	M	X	P	Frequency SRVs Open When Not Needed	Q	Multiplier Frequency
T ₄ MPQ	6E-3		1E-0				.98	5.9E-3
T ₄ MP	6E-3		1E-0		1.3E-6			7.8E-9
T ₄ MQ	6E-3		1E-0			7E-2	2E-2	8.4E-6
T ₄ MQX	6E-3		1E-0	.967		7E-2	2E-2	8.1E-6
T MQX	6E-3		1E-0	3.3E-2		7E-2	2E-2	2.8E-7
T ₄ MPQ	6E-3		1E-0				2E-2	1.2E-4
T ₅ MPQ		3E-3	1E-0				.98	2.9E-3
T ₅ MP		3E-3	1E-0		1.3E-6			3.9E-9
T ₅ MO		3E-3	1E-0			7E-2	2E-2	4.2E-6
T ₅ MQX		3E-3	1E-0	.967		7E-2	2E-2	4.1E-6
T ₅ MQX		3E-3	1E-0	3.3E-2		7E-2	2E-2	1.4E-7
T ₅ MPQ		3E-3	1E-0				2E-2	6.0E-5

Table B.2 ANO 1 Generic Event Data

Fault ID	Fault Description	Failure Prob.	Source
<u>Valves:</u>			
XOC-LF	Manual Valve, NOFC	1E-4	Ref. 1, Item 2.6.2 (1/2) (3E-7/hr)(720 hrs) monthly test interval
XCC-LF	Manual Valve, NCFC	1E-4	Ref. 1, Item 2.6.1
VOC-LF	MOV, NOFC	3.6E-5	Ref. 1, Item 2.1.2 (1/2) (1E-7/hr) (720 hrs) monthly test interval
VOO-LF	MOV, NOFO	3E-3	Ref. 1, Item 2.1.3
VCC-LF	MOV, NCFC	8E-3	Ref. 1, Item 2.1.1 (3E-3) plus control circuit failures (5E-3)
COC-LF	Check Valve, NOFC	3.6E-5	Same as VOC-LF
CCC-LF	Check Valve, NCFC	1E-4	Ref. 1, Item 2.4.1
NOC-LF	Pneumatic-Hyd. Valve NOFC	3.6E-5	Same as VOC-LF
NCC-LF	Pneumatic-Hyd. Valve NCFC	8E-3	Same as VCC-LF
XOC-UTM	Manual Valve not restored after T or M or downtime for T or M	1.1E-4	Ref. 2, Table 20.7, Item 1 fail to restore to proper position after T or M (1E-3)(1/12 monthly check) plus 3E-5 to downtime
XCO-UTM	Same as XOC-UTM	1.1E-4	Same as XOC-UTM
XCC-UTM	Manual Valve unavailability due to downtime	3E-5	Must be opened regardless but could be down for T&M
VOC-UTM	MOV not restored after T or M or downtime for T or M	1.1E-4	Same as XOC-UTM
VOO-UTM	MOV unavailability due to downtime	3E-5	Receives signal to close but could be down for T&M
VCC-UTM	MOV unavailability due to downtime	3E-5	Receives signal to open but could be down for T&M

Table B.2 ANO-1 Generic Event Data (cont)

Fault ID	Fault Description	Failure Prob.	Source
NOC-UTM	Pneumatic-Hyd. Valve NOFC	1.1E-4	Same as XOC-UTM
NCC-UTM	Pneumatic-Hyd. Valve NCFC	1.1E-4	Same as VCC-UTM
SRV-1-LF SRV-2-LF	Pressurizer Safety and Relief Valve, failure to open when pressure exceeds set point (2500 psig)	1E-5	Ref. 1, Item 2.7.1
ERV-LF	Electromatic Relief Valve, failure to open at set point (2450 psig) or when remotely operated	3E-4	Ref. 1, Item 2.9.1
BLKVLV-LF	MOV Block Valve in ERV Line, failure to open	8E-3	Same as VCC-LF
BLKVLV-GATE	This is an artificial gate used in the analysis to permit running certain sequences with and without the ERV Block Valve normally open	1	See Table B.8
RCO-LFM	Normally closed Relief Valve fails open	1E-2	Ref. 7, p. B5-62

Table B-2 ANO-1 Generic Event Data (cont)

Fault ID	Fault Description	Failure Prob.	Source
<u>Pumps:</u>			
PMD-LF	Motor Driven Pump in standby	8.7E-3	Ref. 1, Item 1.1.1 FTS (3E-3) plus Item 1.1.2.1 FTR (3E-5/hr * 24 hr) plus control ckt failures (5E-3)
PMD-LF	Motor Driven Pump normally running (SWS also P4CB-PAC-LF & P4BA-PAC-LF	5.7E-3	Ref. 1, Item 1.1.2.1 for (3E-5/hr * 24 hr) plus control ckt failure (5E-3)
		8.7E-3	For LOSP pumps are stopped and must be restarted
PTD-LF	Turbine Driven Pump in standby	3.5E-2	Ref. 1, Item 1.2.1 FTS (3E-2) plus Item 1.2.2 (1E-5/hr * 24 hr) plus control ckt failure (5E-3)
PMD-UTM	Motor Driven Pump failure to restore to operable state after T or M or downtime for T or M pump normally in standby	2.8E-4	Ref. 2, Table 20.7, Item 1 failure to restore after T or M (1E-3) (1/12 monthly check) plus 2E-4 due to downtime
PMD-UTM	Motor Driven Pump normally running (SWS also PAC-UTM)	0	Given the need for T or M the standby pump will be started.
PTD-UTM	Turbine Driven Pump is standby	6.8E-4	6E-4 downtime plus (1E-3) (1/12 monthly check)

Table B.2 ANO-1 Generic Event Data (cont)

Fault ID	Fault Description	Failure Prob.	Source
<u>Common Mode:</u>			
DG-CM	Diesel Generator	1.5E-3	See text
BAT-CM	Battery	4.4E-4	See text
EFWSP-CM	LF common mode failure between 1PMD and 1PTD	1E-4	See text - ASEP data for EFW pumps specifically
LPSP-CM RBSIP-CM	LF common mode failure between 2 PMD	7E-5	See text - ASEP data
HPSP-CM	LF common mode failure between 3 PMD	3E-5	See text - ASEP data
SWSP-CM	LF common mode failure between 2 PMD normally running and 1 PMD on standby	1E-5	See text - ASEP data
SRV-CM	Pressurizer Safety Relief Valves fail to open due to common mode	1.3E-6	Ref. 1, Item 2.7.1 (1E-5/d) and B factor of 0.13 from the LaSalle PRA
HPSV-CM	Four MOVs in parallel failure to open	1E-4	See text
RBC-SWL-CM	Two trains modeled as trains with two parallel MOVs and two parallel NOV's-failure to open	8E-4	Two times the common mode failure probability for one pair of valves
HPISV-CM LPSV-CM LPRSV-CM RBSIV-CM EFWSV-CM SWSV-08-10-CM SWSV-40-41-CM SWSV-02-03-CM SWSV-21-22-CM SWSV-04-05-CM SWSV-06-07-CM SWSV-40-42-CM	Common mode failure of pairs of MOVs - failure to open	4E-4	See text

Table B.2 ANO-1 Generic Event Data (cont)

Fault ID	Fault Description	Failure Prob.	Source
CSA-CM CSB-CM	CM error to open RBI train recirculation back to BWST - diversion path	4.7E-4	Ref 7, p. B4-47
<u>Buses:</u>			
A1-4160VAC-LF A2-4160VAC-LF B5-480VAC-LF B6-480VAC-LF	AC bus plus two circuit breakers and one transformer	8.4E-5	Ref. 1, Item 4.3.1 (1E-8/hr)(24 hrs) plus Ref. 1, Item 4.1.2 (2*3E-5 spurious trip) plus Ref. 1, Item 4.1.2 (2.4E-5)
A3-4160VAC-LF A4-4160VAC-LF	AC bus plus two circuit breakers	6E-5	Similar to above (1E-8 * 24) + (2 * 3E-5)
480VAC-LF	AC bus plus one circuit breaker	3E-5	Similar to above (1E-8 * 24) + (3E-5)
125VDC-LF LF-DC	DC bus plus one circuit breaker	3E-5	Same as 480VAC-LF
VAC-UTM VDC-UTM TM-DC	T or M on buses and associated components	0	This is consistent with the other TAP A-45 analyses and ASEP
<u>Other Electrical Equipment:</u>			
DG- GEN-LF	Diesel Generator-LF	3.8E-2	See text
DG- GEN-UTM	Diesel Generator-UTM	6.1E-3	Ref. 2, Table 20.7, Item 1 (1E-3)(1/12) plus 6E-3 due to downtime
LF-DC-Y- AC	Inverter AC Power Ckt	1.3E-4	Failure of 1CB (3E-5) + 1 XFMR (2.4E-5) + 1 diode (7.2E-5)
LF-DC-Y	Inverter	2.4E-3	Failure of Inverter (2.4E-3) + 2CB (2 * 3E-5)
LF-DC-RS-ALTAC	Alternate AC Power to RS bus	5.4E-5	Failure of 1 CB (3E-5) + 1 XFMR (2.4E-5)

Table B.2 ANO-1 Generic Event Data (cont)

Fault ID	Fault Description	Failure Prob.	Source
LF-DC-Y-DC	Inverter DC Power Ckt	1.0E-4	Failure of 1CB (3E-5) + 1 DIODE (7.2E-5)
BATTDO- ZBT-LF	Battery-LF	1.8E-3	See text
BATTDO- ZBT-UTM	Battery-UTM	1.1E-3	Ref. 2, Table 20.7, Item 1 (1E-3)(1/12 monthly charger rotation) + 1E-3 due to downtime
DO- CHARGER-LF	Charger	5.4E-5	Failure of Charger 2.4E-5 + Failure of 1CB
CIRCUIT BREAKER (CB)	Spurious Trip	3E-5	Wash 1400 Table 111 2-1 1E-6/hr * 1.25 to convert to mean * 24 hr
BUS	Open Circuit	2.4E-7	Ref. 1, Item 4.3.1 (1E-8/hr) * 24 hrs
CHARGER	Failure to Function	2.4E-5	Ref. 1, Item 4.10.1 (1E-6/hr) * 24 hrs
INVERTER	Failure to Function	2.4E-3	Ref. 1, Item 4.12.1 (1E-4/hr) * 24 hrs
TRANSFORMER (XFMR)	Open Circuit	2.4E-5	Ref. 1, Item 4.5.1 (1E-6/hr) * 24 hrs
CABLE	Open Circuit	7.2E-5	Ref. 1, Item 4.13.1 (3E-6/hr) * 24 hrs
DIODE	Solid State Diode	7.2E-5	Ref. 1, Item 4.14.1 (3E-6/hr) * 24 hrs
LOSS-OSP	Non-initiator loss of offsite power	1E-3	Typical number used in past PRAs
	"	1E-0	Value used for LOSP initiated sequences
TM-LOGIC-CM	CM in T&M or EFWS I&C system logic	2E-4	Ref. 4, 1/10 factor times 2E-3 logic train T&M failures
ACT-SG-SENS-CM	Faults and miscalibrations of analog sensors CM, EFWS	3E-5	Ref. 4

Table B.2 ANO-1 Generic Event Data (cont)

Fault ID	Fault Description	Failure Prob.	Source
ACT-EF--LF	LF in EFWS I&C logic trains	5E-3	Ref. 4 (LF and TM included)
LF-REF-RM	Room cooling refrigeration unit failure	5.5E-3	Ref. 4, p. B13-25
LF-SWS-VCH	Chill water unit cooling failure	1.9E-2	Ref. 4, p. B13-26
LF-ECS-A1	Segment failure in ventilation system	1E-4	Ref. 4, p. B13-25/26
-B1	"	1E-4	"
-A2	"	1E-4	"
-B2	"	1E-4	"
-A4	"	1E-4	"
-B4	"	1E-4	"
LF-ECS-A5	"	3.8E-3	"
-B5	"	4.8E-3	"
-A6	"	4.8E-3	"
-B6	"	6.4E-3	"
<u>Actuation:</u>			
DIG LOG CM-TM	Digital Logic Common Mode	1.7E-6	From LF of digital logic times factor of 0.1
LF-ESF-T	Digital Logic failure	1.7E-5	Ref. 4, p. B9-69
LF-ESF-TRS	Channel Power Supply failure	3.8E-5	Ref. 4, p. B9-70
LOG LF	Analog Logic Unit failure	1E-4	Ref. 4, p. B9-69
LF-ESF-Tb01	Analog Signal path failure with bistable	1.2E-2	Ref. 4, p. B9-71
LF-ESF-TA	Analog bistable failure	1.2E-2	Ref. 4, p. B9-72
LF-ESF-Tb04	Analog signal path failure without bistable	4.5E-4	Ref. 4, p. B9-72
BISTABLEXbb CM	Bistable Common Mode failure	1.2E-3	From LF of bistable times factor of 0.1
ESF-TSL	Applicability gate (external fault tree event)	0	For transients not leading to small LOCAs
		1	For small LOCA or transients leading to small LOCA
			Ref. 4, p. B9-72

Table B.2 ANO-1 Generic Event Data (cont)

Fault ID	Fault Description	Failure Prob.	Source
<u>Other Hardware Faults:</u>			
TNK-LF	Tank Failure	0.0	No failures for tanks
HTX-LFB	Heat Exchanger Local Fault flow blockage	3E-4	Ref. 1, Item 4.4.1 Used number for orifice plugged
HTX-UTM	Heat Exchanger	6E-4	Due to potential downtime
LF-RBC-SEG	Fault in RBCS Segment with 2 dampers & 1 fan	5.9E-2	Ref. 7, p. B6-20 5.4E-2 plus 5E-3 possible downtime
LF-RBC-SWL	Fault in RBCS Segment with 1 VCC, 1 NCC, and 1 HTX	2.3E-2	Ref. 7, p. B6-23
<u>Operator Errors:</u>			
FNB-MANACT	Failure to manually initiate FNB	3E-3	Ref. 2, Table 20.7, Item 3, short list, no checkoff
OPF-OPEN-FNB	Failure to open ERV and ERV Block Valve if closed	3E-3	Same as above
HPRS-MANACT	Failure to manually initiate HPRS	3E-3	Same as above
RBSR-MANACT	Failure to manually initiate RBSR	3E-3	Same as above
LPRS-MANACT	Failure to manually initiate LPRS	3E-3	Same as above
DG-MANACT	Failure to manually start the DGs given failure of DC power	5E-2	Essentially the same as an RA-7D recovery factor

monthly the new one is always cut in after the old one is cut out so that the battery is carrying the load directly for a short time. In discussions with the TAP A-30 (DC power) people it was estimated that 75% to 95% of all the possible battery failure modes can be detected by the quarterly tests. The remaining faults require a discharge test for detection. Therefore the battery local fault probability is based on 85% of the faults being detected with quarterly tests and 15% by the discharge test which leads to:

$$\begin{aligned} \text{BAT-LF} &= 1/2 * (0.85 * 9.2\text{E-}7/\text{h}) * 2190 \text{ hours} + \\ &\quad 1/2 * (0.15 * 9.2\text{E-}7/\text{h}) * 8760 \text{ hrs/yr} * 1.5 \text{ years} \\ &= 8.56\text{E-}4 + 9.06\text{E-}4 = \underline{1.8\text{E-}3} \end{aligned}$$

3.3.2 Local Control Circuit Failure

There is considerable variation in the potential faults applicable to any component and in the faults which are included in the failure probability data given in different data bases. This is especially so for pumps and motor operated valves. We have typically segregated component failure into local faults (LF) and unavailability due to test and maintenance (UTM). The LF normally include the failure to start or operate on demand and the failure to continue running or remain open. For pumps it includes the pump and driver. For valves it includes the valve and its driver. It is not always clear if the interfacing control circuits are included. Therefore, we propose the following, starting with the NUREG/CR-2728 generic data base (Table 5.1-1).

$$\begin{aligned} \text{MDP-LF} &= 3\text{E-}3/\text{d}(\text{FTS}) + 3\text{E-}5/\text{h}(\text{FTR}) * 24\text{h} + 5\text{E-}3(\text{cc}) = \underline{8.7\text{E-}3} \\ \text{TDP-LF} &= 3\text{E-}2/\text{d}(\text{FTS}) * 1\text{E-}5/\text{h}(\text{FTR}) * 24\text{h} + 5\text{E-}3(\text{cc}) = \underline{3.5\text{E-}2} \\ \text{DDP-LF} &= 1\text{E-}3/\text{d}(\text{FTS}) + 8\text{E-}4/\text{h}(\text{FTR}) * 24\text{h} + 5\text{E-}3(\text{cc}) = \underline{2.5\text{E-}2} \end{aligned}$$

where the control circuit (cc) failure probability is derived as follows for the MDP:

Failure of the control circuit breaker to transfer	3E-3
Table 5.1-1 item 4.1.1	
Faults in the control circuit logic (1/2λT)	1E-3
1/2 * 3E-6 * 720h	
Table 5.1-1 item 4.14	
Open circuit in Cable (1/2λT) 1/2 * 3E-6 * 720h	1E-3
Table 5.1-1 item 4.13.1	
	5E-3

This obviously is not a detailed analysis of the hardware. Furthermore, we have added some control circuit failure probability to all three pump types even though we have some verbal opinions stating that this probability could be higher or lower for the TDP and DDP. Unfortunately arriving at

failure probabilities from past data bases is not a precise science (even though it should be). These numbers compare well to the corresponding numbers derived from the NUREG/CR 3226 (A 44), Table E-1 page 228 which includes control circuits as part of the demand (d) failure, i.e.,

$$\begin{aligned} \text{MDP-LF } 2\text{E-}3/\text{d(FTS)} + 2\text{E-}5/\text{h(FTR)} \times 24 &= 2.5\text{E-}3 \\ \text{TDP-LF } 2\text{E-}2/\text{d(FTS)} + 5\text{E-}5/\text{h(FTR)} \times 24 &= 2.1\text{E-}2 \\ \text{DDP-LF } 2\text{E-}2/\text{d(FTS)} + 1\text{E-}4/\text{h(FTR)} \times 24 &= 2.2\text{E-}2 \end{aligned}$$

Similarly for electrically operated valves (e.g., MOV, SOV) that must change positions, the $5\text{E-}3$ control circuit failure probability would need to be added to the hardware failures already considered to obtain the total LF probability.

3.3.3 Unavailability Due to T&M Downtime

It could be assumed that downtime is negligible since routine maintenance is not performed during operation and failures requiring maintenance or repair were relatively infrequent. Industry wide data seems to contradict this, so we will use downtime unavailabilities representative of the data seen in several studies. This unavailability will be added to the failure to restore probability to obtain the total UTM probability. The downtime unavailabilities per reactor year are:

All valves (except check valves) regardless of position	3E-5
Motor driven pump SW & CCS	2E-4
Motor driven pump, other	2E-4
Turbine driven pump	6E-4
Diesel driven pump	1E-3
Heat exchanger	6E-4
Containment fans	5E-3
Battery	1E-3
Diesel generator	6E-3

3.3.4 Common Mode Failures

Diesel generator

The following is based on information found in NUREG/CR-1032, pp 23-15:

$$\text{DG-FTS} = \frac{272 \text{ Failures}}{14304 \text{ Demands}} = 1.9\text{E-}2$$

$$\text{DG-FTR} = (2.4\text{E-}3/\text{h} \times 8\text{h}) = 2.9\text{E-}2$$

$$\begin{aligned} \text{DG-LF} &= \text{DG-FTS} + \text{DG-FTR} \\ &= 1.9\text{E-}2 + 1.9\text{E-}2 = \underline{3.8\text{E-}2} \end{aligned}$$

$$\text{DG-CM factor} = \frac{11 \text{ Identified CM Failures}}{272 \text{ DG Failures}} = 4.0\text{E-}2$$

$$\begin{aligned} \text{DG-CM} &= \text{DG-LF} * \text{DG-CM factor} \\ &= 3.8\text{E-}2 * 4.0\text{E-}2 = \underline{1.5\text{E-}3} \end{aligned}$$

Batteries

The following calculations are based on information found in NUREG/CR-0666 (TAP A-30):

8 failures in 996 battery years (page E-6)

$$\lambda = \frac{8 \text{ Failures}}{996 \text{ Years} * 8760 \text{ hr/yr}} = 9.12\text{E-}7/\text{h}$$

BAT-LF = 3.9E-3 based on the 9.2E-7/h failure rate as discussed previously. For common mode consider (page E-7):

1 common mode occurrence in 996 battery years or

498 battery system years (2 batteries/system)

$$\lambda_{\text{CM}} = \frac{1 \text{ CM Failure}}{498 \text{ BAT SYS YRS}} = 2.0\text{E-}3/\text{yr}$$

Using the same approach used for the LF calculation

$$\begin{aligned} \text{BAT-CM} &= 1/2 * (0.85 * 2.0\text{E-}3/\text{yr}) * 0.25 \text{ yr} + \\ &\quad 1/2 * (0.15 * 2.0\text{E-}3/\text{yr}) * 1.5 \text{ yr} = \\ &\quad 2.13\text{E-}4 + 2.25\text{E-}4 = \underline{4.4\text{E-}4} \end{aligned}$$

Pumps

One approach to the common mode failure of the two EFWS pumps at ANO-1 (1 turbine and 1 motor) uses a 0.06 beta factor derived from the Midland and LaSalle PRAs and an "average" pump failure probability where the pump failures include failure to start, failure to continue operating and interfacing control circuits for the motor driven pumps. Using 3.5E-2 for the TDP and 8.7E-3 for the MDP, the common mode probability would be:

$$(6\text{E-}2) * [(3.5\text{E-}2) * (8.7\text{E-}3)]^{1/2} = 1\text{E-}3$$

If the interfacing control circuit probabilities were excluded, this calculation would become

$$(6\text{E-}2) * [(3\text{E-}2) * (3.7\text{E-}3)]^{1/2} = 6.3\text{E-}4$$

The purpose of the above calculations is an attempts to derive the common mode probability from a β factor currently in use and the basic pump failure probabilities. It has been suggested that the ASEP information from the March 21, 1985, memo to the ASEP Senior Consultant Group is more accurate. That is, for EFWS:

2 trains (1MDP & 1TDP)	4E-4 non-staggered testing 1E-4 staggered testing
2 trains (2TDP)	5E-4 non-staggered testing 2E-4 staggered testing
2 trains (mix of pumps not specified)	4E-4 non-staggered testing 2E-4 staggered testing

ANO-1 has 1TDP and 1MDP with staggered testing. Thus the 1E-4 probability seems reasonable and we will use it.

The ANO-1 SWS has 2 pumps normally operating (but which may have to be restarted in some cases) and 1 pump on standby. The common mode failure probability to be used is 1E-5 which is based on the following ASEP common mode probabilities for pumps with staggered testing.

		Non-staggered Testing	Staggered Testing
Alternating pumps	2 PMD	8E-5	4E-5
	3 PMD	3E-5	1E-5
	4 PMD	2E-5	6E-6

Similarly, ASEP data for other standby pumps with non-staggered testing is:

Standby pumps	2 PMD	1E-4	7E-5
	3 PMD	9E-5	3E-5
	4 PMD	4E-5	1E-5

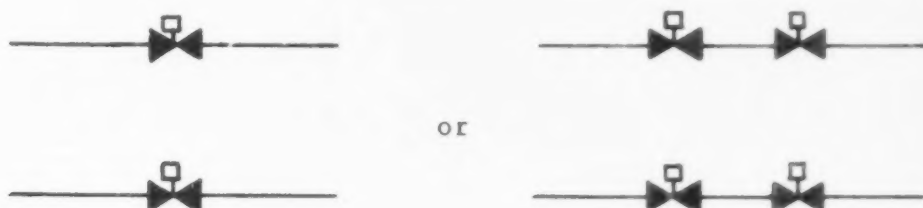
Common mode for other pumps (HP, LP, and CS) is based on the ASEP numbers. The value for the common mode probability for 2 MDPs is 7E-5. For 3 MDPs it is 3E-5.

Motor Operated Valves

Common mode on all pairs of motor operated valves that are normally closed and need to be operated/opened for a system to function will be considered. Manual valve and check valve common mode will not be considered. The local fault failure probability for a MOV is the sum of the failure to open on demand (3E-3) plus the failure of interfacing control circuits (5E-3) which sums to 8E-3. Using a 0.05 beta factor from La

Salle and Midland (actually 0.042) the common mode MOV failure probability is 4E-4.

The 4E-4 common mode failure probability for MOVs will be applied to pairs of MOVs and special cases of 4 MOVs as shown below:



The common mode failure of four parallel valves, as opposed to two pairs of valves in series, is considered to be 1E-4 in our engineering judgment. This configuration is:



The common mode failure of a "relief" valve is derived from a 0.13 beta factor from the La Salle screening estimates leading to the following common mode failure probabilities:

Pressurizer SRV FTO	LF	1E-5	Common Mode	<u>1.3E-6</u>
Secondary ADV FTO or FTOP	LF	3E-4	Common Mode	<u>3.9E-5</u>

4.0 RECOVERY ANALYSIS

The analysis for the application of recovery on the Dominant Accident Sequences (DAS) calculated for ANO-1 is a four step process.

First, the time available for recovery must be determined. This requires an estimate of the time interval between the initiating event and the point at which the core begins to uncover (i.e., the point of no return). This recovery time is based on the accident analysis performed in the Generic Feedwater Studies which provided the basis for the work performed in the Accident Sequence Evaluation Program (ASERP).

Since the ASEP work was concerned with a limited number of sequences the ASEP results were extrapolated to cover all the dominant sequences analyzed for ANO-1 (see Table B.3). However, the thermal hydraulic accident response was not the only thing that affected recovery time. In some specific cases a particular cut set may extend the amount of time available for recovery. For example, a pump may fail due to the absence of room cooling which may take 30 minutes. This extends the recovery time since the actual accident sequence does not start until the pump has failed.

The second step in the recovery process is to identify those failures or basic events which are recoverable. These potentially recoverable failures are examined individually to determine if recovery is possible and by what method it can be accomplished. An entire range of events are considered (see Table B.4) from recovery of offsite power to manually actuating safety injection on a failure of the automatic actuation system. Some events which may at first appear to be recoverable, in reality are not. For example, if a valve on the suction of an HP pump fails to automatically open, the pump could cavitate and irreversibly fail. Manually opening the valve locally would not rectify the failure of the pump which has already occurred.

The third step is to quantify the recovery actions by determining the non-recovery probabilities associated with each action. It is the failure to recover probability which is multiplied with the cut set frequency to determine the reduction in the contributions to core melt. The non-recovery probabilities for electrical failures were taken from those listed in the Station Blackout Accident Analyses NUREC/CR-3226 and expanded in the Accident Sequence Evaluation Program (Ref. 6). The non-recovery probabilities for all other failures are based on the model used in the ANO-1 IREP analysis (Ref. 7).

The final step in the recovery analysis is to re-quantify the DAS. This is performed by computer using the accident sequence in equation form with non-recovery factors included. The Boolean equation is entered into the SETS computer code and new core melt probabilities are calculated.

In the actual application of the recovery methodology, a number of rules were used as listed below.

1. Recovery of offsite power (RA 1) is applied to all T_1 cut sets. Recovery of the PCS (RA 2) applies to the accident sequences shown below.

S_2ML	T_4MLQ
S_3ML	T_4ML
T_2MLQ	T_5MLQ
T_2ML	T_5ML
T_3QL	

Table B.3 ANO-1 Dominant Accident Sequence
Recovery Timing

<u>ANO-1 Generic Accident Sequence</u>	<u>Time Available for Recovery</u>	<u>Source</u>
S ₂ MD, S ₂ MLD	20-30 min (C)	Extrapolated from ASEP TMQD
S ₂ MH, S ₂ MLH	2-4 hours (H)	S ₂ MD + 2 hours
S ₃ ME, S ₃ MLE	40-60 min (E)	Extrapolated from ASEP TMQD
S ₃ MH, S ₃ MLH	2-4 hours (H)	S ₃ MD + 2 hours
TMQD, TMQLD	10-20 min (B)	ASEP TMQD
TMLH, TMQLH	2-4 hours (H)	TMQD + 2 hours
TMLE	30-40 min (D)	ASEP TML

All of the accident sequences fall into one of these categories and the appropriate timing is given in Table B.4.

Table B.4 ANO-1 Non-Recovery Probabilities

		A	B	C	D	E	F	G	H
Recovery of:		5-10 min.	10-20 min.	20-30 min.	30-40 min.	40-60 min.	60-70 min.	70-120 min.	2-4 hrs.
LOSP	RA 1	0.7	0.7	0.5	0.4	0.3	0.3	0.2	0.1
PCS	RA 2	1.0	1.0	1.0	0.9	0.6	0.1	0.03	0.01
Battery CM	RA 8	1.0	1.0	1.0	1.0	1.0	1.0	0.9	0.8
Battery LF	RA 9	1.0	1.0	0.9	0.8	0.7	0.7	0.5	0.2
DG CM	RA 10	1.0	1.0	1.0	1.0	0.9	0.9	0.9	0.7
DG LF	RA 11	1.0	1.0	1.0	1.0	0.9	0.9	0.9	0.8
Other Failure (from Control Room)	RA-6	0.3	0.1	0.05	0.03	0.03	0.01	0.01	0.01
Other Failures (locally)	RA-7	1.0	0.3	0.1	0.05	0.03	0.03	0.01	0.01
Normally Closed Motor Operated Valves (locally)	RA VCC	1.0	0.56	0.49	0.41	0.39	0.39	0.38	0.38

M_L sequences do not require recovery of M. The only T₃ accident sequence that is applicable is T₃QL which is a transient induced LOCA where M does not fail initially but is interrupted when the event becomes a LOCA. In all the T₃M sequences recovery of M is precluded by the 6E-2 failure probability credit given M. RA-1 and RA-2 are never applied to the same cut set.

2. Normally only one recovery action is allowed in addition to either RA-1 or RA-2. Two recovery actions were allowed with either RA-1 or RA-2 for recovery times greater than 2 hours.
3. The CST can be recovered by the SWS if it is available.
4. Cut sets with two UTM events from the same system were considered inconsistent with procedures and thus set to zero.
5. Since only the 2 operating SWS pumps were modeled, it was assumed that an SWS -- PMD-UTM was invalid because the third standby pump would have been substituted. These cut sets were set to zero.
6. None of the cut sets identified in the basecase run were truncated in the recovery analysis.
7. Hardware failures are not directly recoverable, i.e., local faults of pumps and valves cannot be repaired. The exception to this is control circuit failures to a normally closed motor operated valve which could be recovered by a local action to open the valve with the hand wheel. We have combined the standard local fault probability and the control circuit probabilities in this analysis. Thus an additional recovery action (RA-VCC) is defined as follows:

$$RA\ VCC * VCC-LF = VCC-(LF) + VCC-(cc) * RA\ 7$$

$$\text{where } VCC-LF = VCC-(LF) + VCC-(cc)$$

$$VCC-(LF) = 3E-3 \text{ basic local fault failure probability}$$

$$VCC-(cc) = 5E-3 \text{ control circuit failure probability.}$$

The resulting failure to recover probability for RA VCC is given in Table B.4.

8. We will assume that the operator when verifying discharge pressure and flow will observe low values and attempt to regain suction. If suction cannot be reestablished quickly, the pumps will be shut down to prevent cavitation damage. Thus there will be time to manually open valves

if the failure is in the control circuitry and restart the pumps. The following valves should get RA-VCC:

EFWS	Steam inlet	VY-1 & VY-2
FWST	Output	1407 & 1408
HPIS	Discharge	1219 & 1220 & 1227 & 1228
HPIS	RM&LO cooling	3808 & 3810
SUMP	Output	1405 & 1406
LPIS	Discharge	1400 & 1401
LPIS	LO cooling	3840 & 3841
LPIS	RM cooling	3802 & 3803
LPIS	HTX	3821 & 3822
RBSI	Discharge	2400 & 2401
RBSR	LO cooling	3804 & 3805
RBCS	HTX	3814 & 3815
DG	Cooling	3806 & 3807

9. Pump cavitation faults are normally not recoverable. This is accounted for partially in the ANO-1 model by including injection failures that could lead to pump cavitation in the recirculation model. However, suction valve failures can have the recovery action described in item 7 above.
10. The SWS model only includes the 2 normally operating pump trains. The third train can be remotely started and thus provide a backup or recovery action. This applies to PMD or PAC, P4C and P4B, COC 001C and 001B, and XOC 002C and 002B. Since the two diversion paths modeled VOC3820 and 3643 are not diversion flow paths, if two SW pumps are operating, cut sets where a SW pump train fails in combination with the diversion path valve can also be recovered with the standby SW pump. Furthermore, the SWS discharge valve VOC 3824 can also be recovered by opening the valve to the emergency pond. All of these SWS recovery actions are RA-6.
11. SWS failures causing loss of switchgear and battery room cooling failure are recoverable by opening doors and possibly setting up portable fans to circulate the air. There is some doubt as to whether or not the room cooling is even needed for safety purposes. It is thought that the room cooling is to prolong the life of components located in these rooms and that none of the safety related component we have modeled would be degraded in 24 hours. Nevertheless, since these room cooling failures were modeled in IREP and carried over into this analysis, a recovery factor of RA 7G = 0.01 will be applied to

LF-SWS-VCH--
 LF-REF-RM--
 LF-ECS--

since there should be at best one hour of delay time before any possible failures could occur.

12. Actuation failures are recoverable by remote initiation from the control room (RA-6). This applies to the following:

DIG-LOG-CM-TM	
BISTABLEX- -CM	ESAS
LF-ESF- - -	
ESF- - -LOG-LF	
TM-LOGIC-CM	
ACT-SG-SENS-CM	ESFC I&C
ACT-EF- -LF	

13. Common mode failures are normally not considered for recovery. The exception is common mode valve failures which are allowed recovery in the same proportion as normally closed valves discussed in item 7 above. Thus RA-VCC is applicable to the following:

HPSV-CM
HPISV-CM
LPSV-CM
LPRSV-CM
RBSIV-CM
EFWSV-CM
RBC-SWL-CM
SWSV- - -CM except SWSV 40-42-CM

14. The time for recovery can be extended 30 minutes for pump failures due to room cooling. This applies to cut sets where SWS valves CV3802 and 3803 fail LP pump cooling.
15. Successful safety injection was assumed to extend accident sequence timing by approximately three hours over safety injection failure sequences.

5.0 ACCIDENT SEQUENCE ANALYSIS

Once the system success criteria was established and the event trees constructed, the fault tree construction addressed each system or variation on a system that appears in the event tree headings. The data base for quantification was then established for every independent event found in the event trees or fault trees including the containment systems that are part of the separate containment systems event tree. The reason for the separate core melt and containment system event trees is that the potential modifications were based on core melt accident sequences alone, but the value/impact analysis considers risk to the public which involves containment system performance and containment failure modes.

The initial plant visit was to collect information for special emergencies and to discuss numerous internal analysis concerns

such as system success criteria, emergency operating procedures, safety system configurations, and in general any significant changes not documented in the FSAR. This plant visit could have come at any time from near the beginning before the fault trees are started to as late as after the DAS have been established and recovery analyses have been completed. In this case the preliminary event trees, fault trees, and data base were completed before the initial plant visit.

5.1 Accident Sequence Delineation

The first step in the quantification is the delineation of accident sequences. Each LOCA, transient or transient-induced LOCA accident sequence, is identified by a group of symbols representing events that together result in a possible core melt. A condensed version of the event trees found in Sections 2.1 and 2.2 is given in Figure B.7 which shows the connections between the transient and LOCA event trees. The core melt sequences for each initiating event are numbered and designated as late core melt (LCM) or early core melt (ECM). One can trace through the branches of any event tree from the initiating event to end result of success or core melt. The path delineates the success and failure of events (systems and functions) leading to core melt. (Note that we are not interested in the success sequences.) A list of definitions is given in Table B.5.

An example of path from a transient event tree to the LOCA event tree leading to core melt would be $T_3 \overline{MLPQ} \overline{D_1} \overline{P_1} H_1$, i.e., transient event T_3 Sequence Number 4 transferring to LOCA Sequence Number 5. The bar over the symbols means success. In words, this is a transient with offsite power available and the power conversion system initially available (T_3) with subsequent failure of the Power Conversion System (M) and the Auxiliary Feedwater System (L) resulting in the safety relief valves opening successfully ($\overline{P}$) but failing to close (Q) leading to a transient induced LOCA. This LOCA starts on the LOCA tree where the Auxiliary Feedwater System (L) has failed and follows the path where the High Pressure Injection System ($\overline{D_1}$) and Power Operated Relief Valves ($\overline{P_1}$) succeed, that is successful bleed and feed, but leading to a late core melt due to eventual failure of makeup and cooling in the recirculation mode (HPRS with RHR heat exchangers) event H_1 .

Typically, an accident sequence is stated in terms of its failures which, for the example described above, would be $T_3 MLQH_1$, however, the success events must be considered in the quantification procedure. When every possible core melt accident sequence is traced, there are 60 distinct core melt accident sequences.

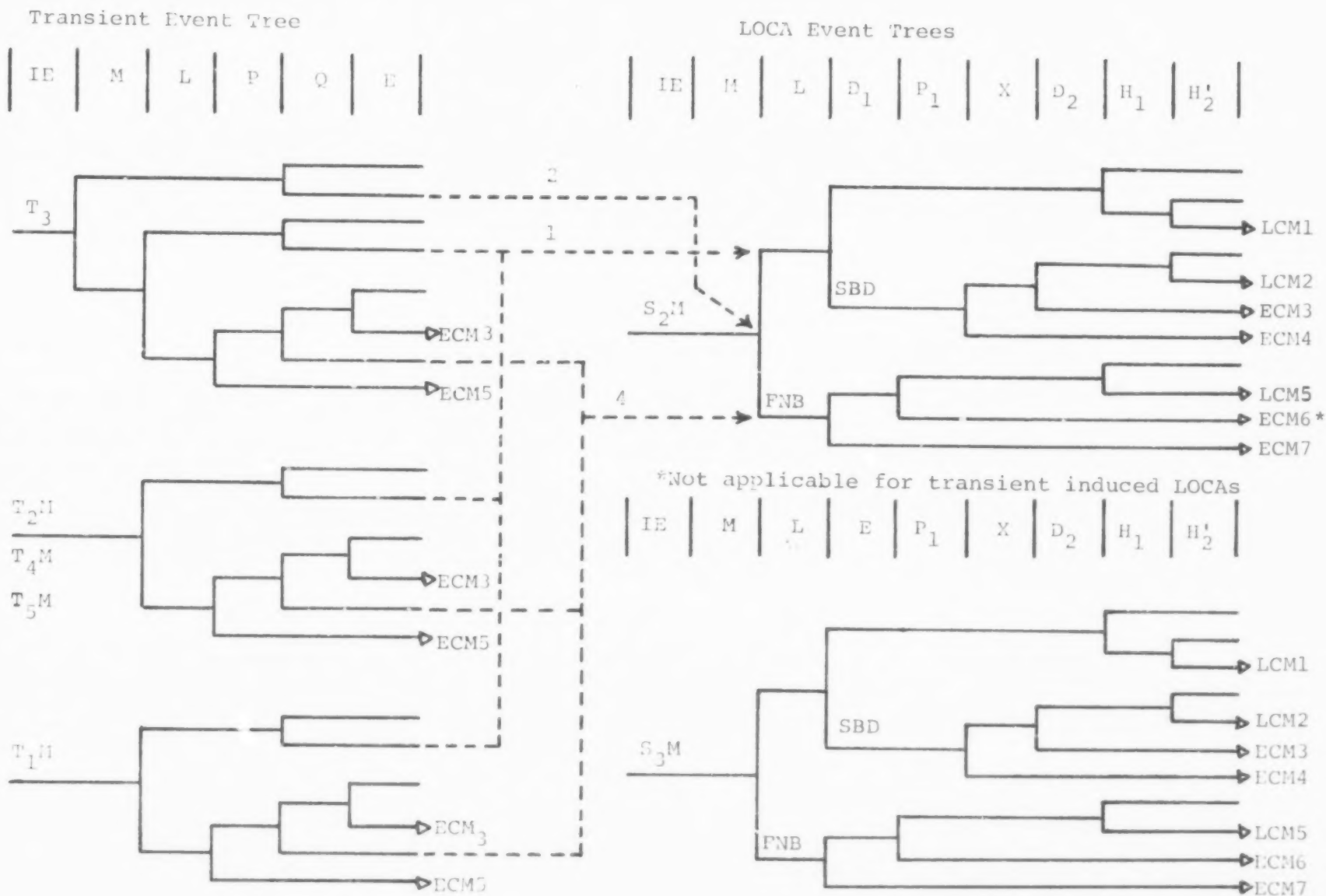


Figure B.7 ANO-1 SDHR System Event Trees

Table B.5 ANO-1 Event Tree/Fault Tree Definitions

T ₁	Loss of offsite power transient
T ₂	Loss of power conversion system transient
T ₃	Transient with PCS initially available
T ₄	Transient loss of AC Bus
T ₅	Transient loss of DC Bus
S ₂	Small LOCA (1.2" - 1.66" diameter break)
S ₃	Small LOCA (0.38" - 1.2" diameter break)
M	Failure of 1 of 2 PCS trains
L	Failure of 1 of 2 EFWS trains
P	Safety or relief valves fail to open
Q	Safety or relief valves fail to close
E	Failure of HPIS when manually initiated
P ₁	Failure to open 1 of 2 SRVs or the ERV
D ₁	Failure of 1 of 3 HPIS trains
D ₂	Failure of 1 of 2 LPIS trains
H ₁	Failure of 1 of 3 HPRS trains with 1 of 2 LPRS trains and 1 of 2 HTXs
H ₂	Failure of 1 of 2 LPRS trains
Y	Failure of 1 of 4 CCUs
C	Failure of 1 of 2 CSIS trains
F	Failure of 1 of 2 CSRS trains

Note: "Failure of" means "failure to achieve success of"

The information given in Table B.6 covers all that is needed to uniquely define each of the 60 accident sequence. The table headings are:

Anal. No.	Analysis number uniquely designating an accident sequence
Ref. No.	Reference number identifying the accident sequence on Figure B.7. An example of a core melt accident sequence reference number is in analysis sequence number 37 where the 45 represents a T ₃ initiating event which becomes a transient induced LOCA (that is the 4) and then follows a LOCA branch to the LOCA core melt outcome labeled 5.
Multiplier	Complete multiplier accident sequence including success and failures. The multiplier events are those events representing systems or functions that are essentially independent of the other events and were not modeled. These events were quantified in Section 3.1.
Boolean Events	Complete Boolean accident sequence including success and failures. The Boolean events are those events for which fault trees were constructed and must be analyzed using Boolean reduction techniques, namely, the SETS computer program.
ANO No.	A number for the ANO-1 analysis that designates a unique set of Boolean events and corresponding value block (VB).
SWS Mod.	There are two different SWS models depending on whether the accident sequence is a transient or a LOCA (including transient induced LOCAs). This affects all the systems that use service water.
VB No.	Value block number designating the changes to the basic value block needed to adapt it to a particular initiating event (see Table B.7).
ESP TSL	The value of the "house event" called ESP-TSL which controls some actuation logic related to LOCAs or transients: 1 for LOCAs and 0 for transients.
Mult. Freq.	Multiplier frequency as described in section 3.1 for the particular accident sequence.
Rec. Time	Recovery time for that accident sequence as described in Section 4.0.
Seq. RA	This designates the applicability of the two general recovery actions (RA) for the entire sequence (i.e., all cut sets).

(Text continued on page B-56.)

Table B.6 ANO-1 Core Melt Accident Sequences

Anal. No.	Ref. No.	Multiplier	Boolean Events	ANO No.	SWS Mod.	VB No.	ESF TSL	Mult. Freq.	Rec. Time	Seq. RA	Core Melt Failure Sequence
1	1	$S_2 M$	$\bar{L} \bar{D}_1 H_1 H'_2$	1	1	1	1	$3.1E-4$	H	-	$S_2 M H_1 H'_2$
2	2	$S_2 M \bar{X}$	$\bar{L} D_1 \bar{D}_2 H'_2$	2	1	1	1	$3.0E-4$	H	-	$S_2 M D_1 H'_2$
3	3	$S_2 M \bar{X}$	$\bar{L} D_1 D_2$	3	1	1	1	$3.0E-4$	C	-	$S_2 M D_1 D_2$
4	4	$S_2 M \bar{X}$	$\bar{L} D_1$	4	1	1	1	$1.0E-5$	C	-	$S_2 M X D_1$
5	5	$S_2 M$	$L \bar{D}_1 \bar{P}_1 H_1$	5	1	1	1	$3.1E-4$	H	2	$S_2 M L H_1$
6	6	$S_2 M$	$L \bar{D}_1 P_1$	6	1	1	1	$3.1E-4$	-	2	$S_2 M L P_1$
7	7	$S_2 M$	$L D_1$	7	1	1	1	$3.1E-4$	C	2	$S_2 M L D_1$
8	1	$S_3 M$	$\bar{L} \bar{E} H_1 H'_2$	9	1	1	1	$2.0E-2$	H	-	$S_3 M H_1 H'_2$
9	2	$S_3 M \bar{X}$	$\bar{L} \bar{E} \bar{D}_2 H'_2$	10	1	1	1	$1.9E-2$	H	-	$S_3 M E H'_2$
10	3	$S_3 M \bar{X}$	$\bar{L} \bar{E} D_2$	11	1	1	1	$1.9E-2$	E	-	$S_3 M E D_2$
11	4	$S_3 M \bar{X}$	$\bar{L} \bar{E}$	12	1	1	1	$6.6E-4$	E	-	$S_3 M X E$
12	5	$S_3 M$	$\bar{L} \bar{E} \bar{P}_1 H_1$	13	1	1	1	$2.0E-2$	H	2	$S_3 M L H_1$
13	6	$S_3 M$	$\bar{L} \bar{E} P_1$	14	1	1	1	$2.0E-2$	-	2	$S_3 M L P_1$
14	7	$S_3 M$	$\bar{L} \bar{E}$	15	1	1	1	$2.0E-2$	E	2	$S_3 M L E$

Table B.6 ANO-1 Core Melt Accident Sequences (continued)

Anal. No.	Ref. No.	Multiplier	Boolean Events	ANO No.	SWS Mod.	VB No.	ESF TSL	Mult. Freq.	Rec. Time	Seq. RA	Core Melt Failure Sequence
15	3	$T_1 \overline{M} \overline{P} \overline{Q}$	LE	22	2	2	0	8.4E-2	D	1	$T_1 \text{MLE}$
16	5	$T_1 \text{MP}$	L	23	2	2	0	1.1E-7	-	1	$T_1 \text{MLP}$
17	3	$T_2 \overline{M} \overline{P} \overline{Q}$	LE	30	2	3	0	9.8E-1	D	2	$T_2 \text{MLE}$
18	5	$T_2 \text{MP}$	L	31	2	3	0	1.3E-6	-	2	$T_2 \text{MLP}$
19	3	$T_3 \overline{M} \overline{P} \overline{Q}$	LE	30a	2	3	0	4.2E-1	D	-	$T_3 \text{MLE}$
20	5	$T_3 \text{MP}$	L	31	2	3	0	5.5E-7	-	-	$T_3 \text{MLP}$
21	11	$T_1 \text{MQ}$	$\overline{L} \overline{D}_1 H_1 H'_2$	16	1	2	1	1.2E-4	H	1	$T_1 \text{MQH}_1 H'_2$
22	12	$T_1 \text{MQX}$	$\overline{L} D_1 \overline{D}_2 H'_2$	17	1	2	1	1.2E-4	H	1	$T_1 \text{MQD}_1 H'_2$
23	13	$T_1 \text{MQX}$	$\overline{L} D_1 D_2$	18	1	2	1	1.2E-4	B	1	$T_1 \text{MQD}_1 D_2$
24	14	$T_1 \text{MQX}$	$\overline{L} D_1$	19	1	2	1	1.2E-4	B	1	$T_1 \text{MQXD}_1$
25	45	$T_1 \overline{M} \overline{P} \overline{Q}$	$L \overline{D}_1 \overline{P}_1 H_1$	20	1	2	1	1.7E-3	H	1	$T_1 \text{QLH}_1$
26	47	$T_1 \overline{M} \overline{P} \overline{Q}$	$L D_1$	21	1	2	1	1.7E-3	B	1	$T_1 \text{QLD}_1$
27	11	$T_2 \text{MQ}$	$\overline{L} \overline{D}_1 H_1 H'_2$	1	1	3	1	1.4E-3	H	-	$T_2 \text{MQH}_1 H'_2$
28	12	$T_2 \text{MQX}$	$\overline{L} D_1 \overline{D}_2 H'_2$	2	1	3	1	1.4E-3	H	-	$T_2 \text{MQD}_1 H'_2$
29	13	$T_2 \text{MQX}$	$\overline{L} D_1 D_2$	3a	1	3	1	1.4E-3	B	-	$T_2 \text{MQD}_1 D_2$
30	14	$T_2 \text{MQX}$	$\overline{L} D_1$	4a	1	3	1	4.6E-5	B	-	$T_2 \text{MQXD}_1$

Table B.6 ANO-1 Core Melt Accident Sequences (continued)

Anal. No.	Ref. No.	Multiplier	Boolean Events	ANO No.	SWS Mod.	VB No.	ESF TSL	Mult. Freq.	Rec. Time	Seq. RA	Core Melt Failure Sequence
31	45	$T_2 \overline{M} \overline{P} Q$	$L \overline{D}_1 \overline{P}_1 H_1$	5	1	3	1	2.0E-2	H	2	$T_2 \overline{M} Q L H_1$
32	47	$T_2 \overline{M} \overline{P} Q$	$L D_1$	7a	1	3	1	2.0E-2	B	2	$T_2 \overline{M} Q L D_1$
33	11	$T_3 \overline{M} Q$	$\overline{L} \overline{D}_1 H_1 H'_2$	1	1	3	1	6.0E-4	H	-	$T_3 \overline{M} Q H_1 H'_2$
34	12	$T_3 \overline{M} Q \overline{X}$	$\overline{L} D_1 \overline{D}_2 H'_2$	2	1	3	1	5.8E-4	H	-	$T_3 \overline{M} Q D_1 H'_2$
35	13	$T_3 \overline{M} Q \overline{X}$	$\overline{L} D_1 D_2$	3a	1	3	1	5.8E-4	B	-	$T_3 \overline{M} Q D_1 D_2$
36	14	$T_3 \overline{M} Q \overline{X}$	$\overline{L} D_1$	4a	1	3	1	2.0E-5	B	-	$T_3 \overline{M} Q X D_1$
37	45	$T_3 \overline{M} \overline{P} Q$	$L \overline{D}_1 \overline{P}_1 H_1$	5a	1	3	1	8.5E-3	H	-	$T_3 \overline{M} Q L H_1$
38	47	$T_3 \overline{M} \overline{P} Q$	$L D_1$	7b	1	3	1	8.5E-3	B	-	$T_3 \overline{M} Q L D_1$
39	21	$T_3 Q$	$\overline{L} \overline{D}_1 H_1 H'_2$	1	1	3	1	9.9E-3	H	-	$T_3 Q H_1 H'_2$
40	22	$T_3 Q \overline{X}$	$\overline{L} D_1 \overline{D}_2 H'_2$	2	1	3	1	9.6E-3	H	-	$T_3 Q D_1 H'_2$
41	23	$T_3 Q \overline{X}$	$\overline{L} D_1 D_2$	3a	1	3	1	9.6E-3	B	-	$T_3 Q D_1 D_2$
42	24	$T_3 Q \overline{X}$	$\overline{L} D_1$	4a	1	3	1	3.3E-4	B	-	$T_3 Q X D_1$
43	25	$T_3 Q$	$L \overline{D}_1 \overline{P}_1 H_1$	5	1	3	1	9.9E-3	H	2	$T_3 Q L H_1$
44	27	$T_3 Q$	$L D_1$	7a	1	3	1	9.9E-3	B	2	$T_3 Q L D_1$

Table B.6 ANO-1 Core Melt Accident Sequences (continued)

Anal. No.	Ref. No.	Multiplier	Boolean Events	ANO No.	SWS Mod.	VB No.	ESF TSL	Mult. Freq.	Rec. Time	Seq. RA	Core Melt Failure Sequence
45	11	$T_4 MQ$	$\bar{L} \bar{D}_1 H_1 H'_2$	32	1	4	1	8.4E-6	H	-	$T_4 MQH_1 H'_2$
46	12	$T_4 MQ\bar{X}$	$\bar{L} D_1 \bar{D}_2 H'_2$	33	1	4	1	8.1E-6	H	-	$T_4 MQD_1 H'_2$
47	13	$T_4 MQ\bar{X}$	$\bar{L} D_1 D_2$	34	1	4	1	8.1E-6	B	-	$T_4 MQD_1 D_2$
48	14	$T_4 MQX$	$\bar{L} D_1$	35	1	4	1	2.8E-7	B	-	$T_4 MQXD_1$
49	45	$T_4 MPQ$	$L \bar{D}_1 \bar{P}_1 H_1$	36	1	4	1	1.2E-4	H	2	$T_4 MQLH_1$
50	47	$T_4 MPQ$	$L D_1$	37	1	4	1	1.2E-4	B	2	$T_4 MQLD_1$
51	3	$T_4 MP\bar{Q}$	LE	38	2	4	0	5.9E-3	D	2	$T_4 MLE$
52	5	$T_4 MP$	L	39	2	4	0	7.8E-9	-	2	$T_4 MLP$
53	11	$T_5 MQ$	$\bar{L} \bar{D}_1 H_1 H'_2$	40	1	5	1	4.2E-6	H	-	$T_5 MQH_1 H'_2$
54	12	$T_5 MQ\bar{X}$	$\bar{L} D_1 \bar{D}_2 H'_2$	41	1	5	1	4.1E-6	H	-	$T_5 MQD_1 H'_2$
55	13	$T_5 MQ\bar{X}$	$\bar{L} D_1 D_2$	42	1	5	1	4.1E-6	B	-	$T_5 MQD_1 D_2$
56	14	$T_5 MQX$	$\bar{L} D_1$	43	1	5	1	1.4E-7	B	-	$T_5 MQXD_1$
57	45	$T_5 MPQ$	$L \bar{D}_1 \bar{P}_1 H_1$	44	1	5	1	6.0E-5	H	2	$T_5 MQLH_1$
58	47	$T_5 MPQ$	$L D_1$	45	1	5	1	6.0E-5	B	2	$T_5 MQLD_1$
59	3	$T_5 MP\bar{Q}$	LE	46	2	5	0	2.9E-3	D	2	$T_5 MLE$
60	5	$T_5 MP$	L	47	2	5	0	3.9E-9	-	2	$T_5 MLP$

Table B.7 Value Block Changes for Accident Sequence

Runs Change Value Block/Input as Follows:

<u>Event</u>	<u>VB-1</u>	<u>VB-2</u>	<u>VB-3</u>	<u>VB-4</u>	<u>VB-5</u>
LOSS-OSP	1.0E-3	OMEGA	1.0E-3	1.0E-3	1.0E-3
SWSOP4BA-PAC-LF	5.7E-3	8.7E-3(NC)	5.7E-3	5.7E-3	5.7E-3
SWSOP4BA-PMD-LF	5.7E-3	8.7E-3(NC)	5.7E-3	5.7E-3	5.7E-3
SWSOP4CB-PAC-LF	5.7E-3	8.7E-3(NC)	5.7E-3	5.7E-3	5.7E-3
SWSOP4CB-PMD-LF	5.7E-3	8.7E-3(NC)	5.7E-3	5.7E-3	5.7E-3
A3-4160VAC-LF	6.0E-5	6.0E-5	6.0E-5	OMEGA	6.0E-5
D01-125VDC-LF	3.0E-5	3.0E-5	3.0E-5	3.0E-5	OMEGA
BLKVLV-LF	8.0E-3	8.0E-3	8.0E-3	8.0E-3	8.0E-3
BLKVLV-GATE	OMEGA	OMEGA	OMEGA	OMEGA	OMEGA
ESF-TSL LOCA & Transient Induced LOCA	OMEGA	OMEGA	OMEGA	OMEGA	OMEGA
Transient	PH1	PH1	PH1	PH1	PH1
	<u>VB-1a</u>	<u>VB-2a</u>	<u>VB-3a</u>	<u>VB-4a</u>	<u>VB-5a</u>
LOSS-OSP	1.0E-3	OMEGA	1.0E-3	1.0E-3	1.0E-3
SWSOP4BA-PAC-LF	5.7E-3	8.7E-3(NC)	5.7E-3	5.7E-3	5.7E-3
SWSOP4BA-PMD-LF	5.7E-3	8.7E-3(NC)	5.7E-3	5.7E-3	5.7E-3
SWSOP4CB-PAC-LF	5.7E-3	8.7E-3(NC)	5.7E-3	5.7E-3	5.7E-3
SWSOP4CB-PMD-LF	5.7E-3	8.7E-3(NC)	5.7E-3	5.7E-3	5.7E-3
A3-4160VAC-LF	6.0E-5	6.0E-5	6.0E-5	OMEGA	6.0E-5
D01-125VDC-LF	3.0E-5	3.0E-5	3.0E-5	3.0E-5	OMEGA
BLKVLV-LF	3.6E-5	3.6E-5	3.6E-5	3.6E-5	3.6E-5
BLKVLV-GATE	PH1	PH1	PH1	PH1	PH1
ESF-TSL LOCA & Transient Induced LOCA	OMEGA	OMEGA	OMEGA	OMEGA	OMEGA
Transient	PH1	PH1	PH1	PH1	PH1

Core Melt Failure Sequence = Typically an accident sequence is given in terms of just the failure events leading to core melt.

Table B.8 delineates the Boolean sequences to be run by SETS and numbered ANO No. 1 through 58 including each unique sequence, applicable value block, SWS model (there are 2 models depending on the initiating event), and ESF TSL (a logic event in the ESAS model that depends on the initiating event) value. The Boolean events are shown spread out so the computer analyst can distinguish patterns to aid in constituting the most efficient SETS runs. Some of the sequences are only used in sensitivity analyses and are not part of the base case. As a point of reference, the SETS code is used in this program but will not be discussed here since it is well documented elsewhere.

When the fault trees for all the systems are initially brought together, a value block is used that does not distinguish between the initiating events. The value block is a listing of every basic event by its SETS code name and the assigned probability. When the accident sequences are run, minor adjustments are made to make the value block to fit the initiating event. These adjustments are given in Table B.7.

5.2 Accident Sequence Quantification

It was assumed in the event trees shown in Figure B.7 that the feed and bleed mode and the secondary blowdown mode were applicable at ANO-1. The first subsection describes the quantification of the basecase event trees as shown in Figure B.7. The next two subsections address the quantification of the event trees if feed and bleed or secondary blowdown had not been applicable. This will show the apparent value of these two modes of operation. The last subsection addresses the effect of the ERV block valve being unblocked (open) during normal operations.

5.2.1 Quantification Including Feed and Bleed, and Secondary Blowdown

The results of the SETS runs are given in Table B.9 using a $1E-7$ truncation and before recovery was applied. The core melt probabilities are computed by the product of the multiplier frequency and the Boolean sequence probability. Note that several Boolean sequences, designated by the ANO number, are used with more than one multiplier. All of the accident sequences marked with an * were carried forward into the recovery analysis. This amounted to 23 sequences which accounted for 99% of the core melt probability. The 23 sequences are given in Table B.10.

The recovery analysis methods are given in Section 4.0. After the recovery factors were applied to the top 23 accident sequences, the core melt probability was reduced from $4.66E-4$ to $8.48E-5$. The core melt probabilities are given in Table B.11

(Text continued on page B-70.)

Table B.6 ANO-1 Boolean Accident Sequences

ANO No.	L	E	D ₁	P ₁	D ₂	H ₁	H' ₂	VB	SWS MOD.	ESF TSL
1	$\bar{L}$		$\bar{D}_1$			H ₁	H' ₂	1	1	1
2	$\bar{L}$		D ₁		$\bar{D}_2$		H' ₂	1	1	1
3	$\bar{L}$		D ₁		D ₂			1	1	1
4	$\bar{L}$		D ₁					1	1	1
5	L		$\bar{D}_1$	$\bar{P}_1$		H ₁		1	1	1
6	L		$\bar{D}_1$	P ₁				1	1	1
7	L		D ₁					1	1	1
8	L							1	1	1
9	$\bar{L}$	$\bar{E}$				H ₁	H' ₂	1	1	1
10	$\bar{L}$	E			$\bar{D}_2$		H' ₂	1	1	1
11	$\bar{L}$	E			D ₂			1	1	1
12	$\bar{L}$	E						1	1	1
13	L	$\bar{E}$		$\bar{P}_1$		H ₁		1	1	1
14	L	$\bar{E}$		P				1	1	1
15	L	E						1	1	1
16	$\bar{L}$		$\bar{D}_1$			H ₁	H' ₂	2	1	1
17	$\bar{L}$		D ₁		$\bar{D}_2$		H' ₂	2	1	1
18	$\bar{L}$		D ₁		D ₂			2	1	1
19	$\bar{L}$		D ₁					2	1	1
20	L		$\bar{D}_1$	$\bar{P}_1$		H ₁		2	1	1

Table B.8 ANO-1 Boolean Accident Sequences (cont)

ANO No.	L	E	D ₁	P ₁	D ₂	H ₁	H ₂	VB	SWS MOD.	ESF TSL
21	L		D ₁					2	1	1
22	L	E						2	2	0
23	L							2	2	0
24	$\bar{L}$		$\bar{D}_1$			H ₁	H ₂	3	Identical to Boolean Accident Sequence 1-5, 7 since VB1 = VB3 for ANO-1	
25	$\bar{L}$		D ₁		$\bar{D}_2$		H ₂	3		
26	$\bar{L}$		D ₁		D ₂			3		
27	$\bar{L}$		D ₁					3		
28	L		$\bar{D}_1$	$\bar{P}_1$		H ₁		3		
29	L		D ₁					3		
30	L	E						3	2	0
31	L							3	2	0
32	$\bar{L}$		$\bar{D}_1$			H ₁	H ₂	4	1	1
33	$\bar{L}$		D ₁		$\bar{D}_2$		H ₂	4	1	1
34	$\bar{L}$		D ₁		D ₂			4	1	1
35	$\bar{L}$		D ₁					4	1	1
36	L		$\bar{D}_1$	$\bar{P}_1$		H ₁		4	1	1
37	L		D ₁					4	1	1
38	L	E						4	2	0
39	L							4	2	0

Table B.8 ANO-1 Boolean Accident Sequences (cont)

ANO No.	L	E	D_1	P_1	D_2	H_1	H'_2	VB	SWS MOD.	ESF TSL
40	$\bar{L}$		$\bar{D}_1$			H_1	H'_2	5	1	1
41	$\bar{L}$		D_1		$\bar{D}_2$		H'_2	5	1	1
42	$\bar{L}$		D_1		D_2			5	1	1
43	$\bar{L}$		D_1					5	1	1
44	L		$\bar{D}_1$	$\bar{P}_1$		H_1		5	1	1
45	L		D_1					5	1	1
46	L	E						5	2	0
47	L							5	2	0
48	L		$\bar{D}_1$	$\bar{P}_1$		H_1		1a	1	1
49	L		$\bar{D}_1$	P_1				1a	1	1
50	L	$\bar{E}$		$\bar{P}_1$		H_1		1a	1	1
51	L	$\bar{E}$		P_1				1a	1	1
52	L		$\bar{D}_1$	$\bar{P}_1$		H_1		2a	1	1
53	L		$\bar{D}_1$	$\bar{P}_1$		H_1		3a	Identical to 48	
54	L		$\bar{D}_1$	$\bar{P}_1$		H_1		4a	1	1
55	L		$\bar{D}_1$	$\bar{P}_1$		H_1		5a	1	1
56	L							2	1	1
57	L							4	1	1
58	L							5	1	1

Table B.9 ANO-1 Accident Sequence Core Melt Probability
Before Recovery (10^{-7} Truncation)

Anal. No.	ANO No.	Mult. Freq.	Rec.* Time	Seq.** RA	Boolean Sequence Prob.	Acc.Seq. Core Melt Prob.	Core Melt Failure Sequence
1	1	3.1E-4	H	-	5.77E-3	1.79E-6*	S ₂ MH ₁ H' ₂
2	2	3.0E-4	H	-	1.57E-5	4.71E-9	S ₂ MD ₁ H' ₂
3	3	3.0E-4	C	-	2.85E-3	8.55E-7	S ₂ MD ₁ D ₂
4	4	1.0E-5	C	-	3.89E-3	3.89E-8	S ₂ MXD ₁
5	5	3.1E-4	H	2	5.74E-5	1.78E-8	S ₂ MLH ₁
6	6	3.1E-4	-	2	Φ	Φ	S ₂ MLP ₁
7	7	3.1E-4	C	2	7.54E-5	2.34E-8	S ₂ MLD ₁
8	9	2.0E-2	H	-	5.77E-3	1.15E-4*	S ₃ MH ₁ H' ₂
9	10	1.9E-2	H	-	3.57E-5	6.78E-7	S ₃ MEH' ₂
10	11	1.9E-2	E	-	2.86E-3	5.43E-5*	S ₃ MED ₂
11	12	6.6E-4	E	-	6.89E-3	4.55E-6*	S ₃ MXE
12	13	2.0E-2	H	2	5.74E-5	1.15E-6*	S ₃ MLH ₁
13	14	2.0E-2	-	2	Φ	Φ	S ₃ MLP ₁
14	15	2.0E-2	E	2	8.11E-5	1.62E-6*	S ₃ MLE

* Recovery time - see Tables B3 and B.4.

** Sequence Recovery Action applies to all cut sets in the
sequence - see Table B.4.

Table B.9 ANO-1 Accident Sequence Core Melt Probability
Before Recovery (10^{-7} Truncation) (cont)

Anal. No.	ANO No.	Mult. Freq.	Rec. Time	Seq. RA	Boolean Sequence Prob.	Acc.Seq. Core Melt Prob.	Core Melt Failure Sequence
15	22	8.4E-2	D	1	8.65E-4	7.27E-5*	T ₁ MLE
16	23	1.1E-7	-	1	4.31E-3	4.74E-10	T ₁ MLP
17	30	9.8E-1	D	2	5.77E-5	5.65E-5*	T ₂ MLE
18	31	1.3E-6	-	2	1.74E-3	2.26E-9	T ₂ MLP
19	30a	4.2E-1	D	-	5.77E-5	2.42E-5*	T ₃ MLE
20	31	5.5E-7	-	-	1.74E-3	9.57E-10	T ₃ MLP
21	16	1.2E-4	H	1	1.00E-2	1.20E-6*	T ₁ MQH ₁ H' ₂
22	17	Zero	H	1	3.57E-5	φ	T ₁ MQD ₁ H' ₂
23	18	Zero	B	1	1.44E-2	φ	T ₁ MQD ₁ D ₂
24	19	1.2E-4	B	1	1.65E-2	1.98E-6*	T ₁ MQXD ₁
25	20	1.7E-3	H	1	1.56E-4	2.65E-7	T ₁ MQLH ₁
26	21	1.7E-3	B	1	9.29E-4	1.58E-6*	T ₁ MQLD ₁
27	1	1.4E-3	H	-	5.77E-3	8.07E-6*	T ₂ MQH ₁ H' ₂
28	2	1.4E-3	H	-	1.57E-5	2.20E-8	T ₂ MQD ₁ H' ₂
29	3a	1.4E-3	B	-	2.85E-3	3.99E-6*	T ₂ MQD ₁ D ₂
30	4a	4.6E-5	B	-	3.89E-3	1.79E-7	T ₂ MQXD ₁

Table B.9 ANO-1 Accident Sequence Core Melt Probability
Before Recovery (10^{-7} Truncation) (cont)

Anal. No.	ANO No.	Mult. Freq.	Rec. Time	Seq. RA	Boolean Sequence Prob.	Acc.Seq. Core Melt Prob.	Core Melt Failure Sequence
31	5	2.0E-2	H	2	5.74E-5	1.15E-6*	T ₂ MQLH ₁
32	7a	2.0E-2	B	2	7.54E-5	1.51E-6*	T ₂ MQLD ₁
33	1	6.0E-4	H	-	5.77E-3	3.46E-6*	T ₃ MQH ₁ H' ₂
34	2	5.8E-4	H	-	1.57E-5	9.11E-9	T ₃ MQD ₁ H' ₂
35	3a	5.8E-4	B	-	2.85E-3	1.65E-6*	T ₃ MQD ₁ D ₂
36	4a	2.0E-5	B	-	3.89E-3	7.78E-8	T ₃ MQXD ₁
37	5a	8.5E-3	H	-	5.74E-5	4.88E-7	T ₃ MQLH ₁
38	7b	8.5E-3	B	-	7.54E-5	6.41E-7	T ₃ MQLD ₁
39	1	9.9E-3	H	-	5.77E-3	5.71E-5*	T ₃ QH ₁ H' ₂
40	2	9.6E-3	H	-	1.57E-5	1.51E-7	T ₃ QD ₁ H' ₂
41	3a	9.6E-3	B	-	2.85E-3	2.74E-5*	T ₃ QD ₁ D ₂
42	4a	3.3E-4	B	-	3.89E-3	1.28E-6*	T ₃ QXD ₁
43	5	9.9E-3	H	2	5.74E-5	5.68E-7	T ₃ QLH ₁
44	7a	9.9E-3	B	2	7.54E-5	7.46E-7	T ₃ QLD ₁

Table B.9 ANO-1 Accident Sequence Core Melt Probability
Before Recovery (10^{-7} Truncation) (cont)

Anal. No.	ANO No.	Mult. Freq.	Rec. Time	Seq. RA	Boolean Sequence Prob.	Acc.Seq. Core Melt Prob.	Core Melt Failure Sequence
45	32	8.4E-6	H	-	3.63E-2	3.05E-7	T ₄ MOH ₁ H ₂ '
46	33	8.1E-6	H	-	3.80E-4	3.08E-9	T ₄ MOD ₁ H ₂ '
47	34	8.1E-6	B	-	4.57E-2	3.70E-7	T ₄ MOD ₁ D ₂
48	35	2.8E-7	B	-	6.38E-2	1.79E-8	T ₄ MOXD ₁
49	36	1.2E-4	H	2	1.80E-3	2.16E-7	T ₄ MOH ₁
50	37	1.2E-4	B	2	2.35E-3	2.82E-7	T ₄ MOD ₁
51	38	5.9E-3	D	2	2.23E-3	1.32E-5*	T ₄ MLE
52	39	7.8E-9	-	2	3.67E-2	2.86E-10	T ₄ MLP
53	40	4.2E-6	H	-	3.63E-2	1.52E-7	T ₅ MOH ₁ H ₂ '
54	41	4.1E-6	H	-	4.11E-4	1.69E-9	T ₅ MOD ₁ H ₂ '
55	42	4.1E-6	B	-	4.52E-2	1.85E-7	T ₅ MOD ₁ D ₂
56	43	1.4E-7	B	-	6.33E-2	8.86E-9	T ₅ MOXD ₁
57	44	6.0E-5	H	2	2.43E-3	1.46E-7	T ₅ MOH ₁
58	45	6.0E-5	B	2	3.55E-3	2.13E-7	T ₅ MOD ₁
59	46	2.9E-3	D	2	3.32E-3	9.64E-6*	T ₅ MLE
60	47	3.9E-9	-	2	4.98E-2	1.94E-10	T ₅ MLP
All Sequences						4.7204E-4	
Top 23 * Sequences only						4.6637E-4	
Next 37 Sequences						5.67E-6	

Table B.10 ANO-1 Results Before Recovery
Using 1E-7 SETS Truncation -
Top 23 Accident Sequences

Ranking	Analysis Number	Core Melt Failure Sequence	Core Melt Probability	Boolean Sequence Number
1	8	$S_3^{MH_1H_2'}$	1.15E-4	9
2	15	T_1^{MLE}	7.27E-5	22
3	39	$T_3^{QH_1H_2'}$	5.71E-5	1
4	17	T_2^{MLE}	5.66E-5	30
5	10	$S_3^{MED_2}$	5.43E-5	11
6	41	$T_3^{QD_1D_2}$	2.74E-5	3a
7	19	T_3^{MLE}	2.42E-5	30a
8	51	T_4^{MLE}	1.32E-5	38
9	59	T_5^{MLE}	9.64E-6	46
10	27	$T_2^{MQH_1H_2'}$	8.07E-6	1
11	11	S_3^{MXE}	4.55E-6	12
12	29	$T_2^{MQD_1D_2}$	3.99E-6	3a
13	33	$T_3^{MQH_1H_2'}$	3.46E-6	1
14	24	$T_1^{MQXD_1}$	1.98E-6	19
15	1	$S_2^{MH_1H_2'}$	1.79E-6	1
16	35	$T_3^{MQD_1D_2}$	1.65E-6	3a
17	14	S_3^{MLE}	1.62E-6	15
18	26	$T_1^{MQLD_1}$	1.58E-6	21
19	32	$T_2^{MQLD_1}$	1.51E-6	7a
20	42	$T_3^{QXD_1}$	1.28E-6	4a
21	21	$T_1^{MQH_1H_2'}$	1.20E-6	16
22	12	$S_3^{MLH_1}$	1.15E-6	13
23	31	$T_2^{MQLH_1}$	1.15E-6	5
			4.6637E-4	

Table B.11 ANO-1 Accident Sequence Core Melt Probability After Recovery (No additional truncation from Table B.9)

Anal. No.	ANO No.	Mult. Freq.	Rec.* Time	Seq.** RA	Boolean Sequence Prob.	Acc.Seq. Core Melt Prob.	Core Melt Failure Sequence
1	1	3.1E-4	H	-	1.22E-3	3.78E-7	S ₂ MH ₁ H ₂ '
2	2	3.0E-4	H	-	NC	4.71E-9	S ₂ MD ₁ H ₂ '
3	3	3.0E-4	C	-	3.96E-4	1.19E-7	S ₂ MD ₁ D ₂
4	4	1.0E-5	C	-	6.13E-4	6.13E-9	S ₂ MXD ₁
5	5	3.1E-4	H	2	2.25E-8	6.98E-12	S ₂ MLH ₁
6	6	3.1E-4	-	2	NC	Φ	S ₂ MLP ₁
7	7	3.1E-4	C	2	1.86E-6	5.77E-10	S ₂ MLD ₁
8	9	2.0E-2	H	-	1.12E-3	2.24E-5*	S ₃ MH ₁ H ₂ '
9	10	1.9E-2	H	-	NC	6.78E-7	S ₃ MEH ₂ '
10	11	1.9E-2	E	-	3.47E-4	6.60E-6*	S ₃ MED ₂
11	12	6.6E-4	E	-	3.52E-3	2.33E-6*	S ₃ MXE
12	13	2.0E-2	H	2	2.01E-8	4.02E-10	S ₃ MLH ₁
13	14	2.0E-2	-	2	NC	Φ	S ₃ MLP ₁
14	15	2.0E-2	E	2	1.75E-6	3.50E-8	S ₃ MLE

* Recovery time - see Tables B.3 and B.4.

** Sequence Recovery Action applies to all cut sets in the sequence - see Table B.4.

Table B.11 ANO-1 Accident Sequence Core Melt Probability
After Recovery (No additional truncation from
Table B.9) (cont.)

Anal. No.	ANO No.	Mult. Freq.	Rec. Time	Seq. RA	Boolean Sequence Prob.	Acc.Seq. Core Melt Prob.	Core Melt Failure Sequence
15	22	8.4E-2	D	1	2.54E-4	2.13E-5*	T ₁ MLE
16	23	1.1E-7	-	1	NC	4.74E-10	T ₁ MLP
17	30	9.8E-1	D	2	2.43E-6	2.38E-6*	T ₂ MLE
18	31	1.3E-6	-	2	NC	2.26E-9	T ₂ MLP
19	30a	4.2E-1	D	-	2.70E-6	1.13E-6*	T ₃ MLE
20	31	5.5E-7	-	-	NC	9.57E-10	T ₃ MLP
21	16	1.2E-4	H	1	3.05E-4	3.66E-8	T ₁ MQH ₁ H ₂ ¹
22	17	Zero	H	1	NC	φ	T ₁ MQD ₁ H ₂ ¹
23	18	Zero	B	1	NC	φ	T ₁ MQD ₁ D ₂
24	19	1.2E-4	B	1	4.95E-3	5.94E-7	T ₁ MQXD ₁
25	20	1.7E-3	H	1	1.56E-5	2.65E-8	T ₁ MQH ₁
26	21	1.7E-3	B	1	4.61E-4	7.84E-7*	T ₁ MQD ₁
27	1	1.4E-3	H	-	1.22E-3	1.71E-6*	T ₂ MQH ₁ H ₂ ¹
28	2	1.4E-3	H	-	NC	2.20E-8	T ₂ MQD ₁ H ₂ ¹
29	3a	1.4E-3	B	-	4.76E-4	6.66E-7*	T ₂ MQD ₁ D ₂
30	4a	4.6E-5	B	-	7.40E-4	3.40E-8	T ₂ MQXD ₁

Table B.11 ANO-1 Accident Sequence Core Melt Probability
After Recovery (No additional truncation from
Table B.9) (cont.)

Anal. No.	ANO No.	Mult. Freq.	Rec. Time	Seq. RA	Boolean Sequence Prob.	Acc.Seq. Core Melt Prob.	Core Melt Failure Sequence
31	5	2.0E-2	H	2	2.25E-8	4.50E-10	T ₂ MQLH ₁
32	7a	2.0E-2	B	2	2.57E-6	5.14E-8	T ₂ MQLD ₁
33	1	6.0E-4	H	-	1.22E-3	7.32E-7*	T ₃ MOH ₁ H ₂ '
34	2	5.8E-4	H	-	NC	9.11E-9	T ₃ MOD ₁ H ₂ '
35	3a	5.8E-4	B	-	4.76E-4	2.76E-7	T ₃ MOD ₁ D ₂
36	4a	2.0E-5	B	-	7.40E-4	1.48E-8	T ₃ MOXD ₁
37	5a	8.5E-3	H	-	2.25E-6	1.92E-8	T ₃ MQLH ₁
38	7b	8.5E-3	B	-	2.57E-6	2.19E-8	T ₃ MQLD ₁
39	1	9.9E-3	H	-	1.22E-3	1.21E-5*	T ₃ QH ₁ H ₂ '
40	2	9.6E-3	H	-	NC	1.51E-7	T ₃ QD ₁ H ₂ '
41	3a	9.6E-3	B	-	4.76E-4	4.57E-6*	T ₃ QD ₁ D ₂
42	4a	3.3E-4	B	-	7.40E-4	2.44E-7	T ₃ QXD ₁
43	5	9.9E-3	H	2	2.25E-8	2.23E-10	T ₃ QLH ₁
44	7a	9.9E-3	B	2	2.57E-6	2.54E-8	T ₃ QLD ₁

Table B.11 ANO-1 Accident Sequence Core Melt Probability
After Recovery (No additional truncation from
Table B.9) (cont.)

Anal. No.	ANO No.	Mult. Freq.	Rec. Time	Seq. RA	Boolean Sequence Prob.	Acc.Seq. Core Melt Prob.	Core Melt Failure Sequence
45	32	8.4E-6	H	-	NC	3.05E-7	$T_4MQH_1H'_2$
46	33	8.1E-6	H	-	NC	3.08E-9	$T_4MQD_1H'_2$
47	34	8.1E-6	B	-	NC	3.70E-7	$T_4MQD_1D_2$
48	35	2.8E-7	B	-	NC	1.79E-8	T_4MQXD_1
49	36	1.2E-4	H	2	1.80E-5	2.16E-9	T_4MQLH_1
50	37	1.2E-4	B	2	NC	2.82E-7	T_4MQLD_1
51	38	5.9E-3	D	2	6.91E-4	4.08E-6*	T_4MLE
52	39	7.8E-9	-	2	NC	2.86E-10	T_4MLP
52	40	4.2E-6	H	-	NC	1.52E-7	$T_5MQH_1H'_2$
54	41	4.1E-6	H	-	NC	1.69E-9	$T_5MQD_1H'_2$
55	42	4.1E-6	B	-	NC	1.85E-7	$T_5MQD_1D_2$
56	43	1.4E-7	B	-	NC	8.86E-9	T_5MQXD_1
57	44	6.0E-5	H	2	2.43E-5	1.46E-9	T_5MQLH_1
58	45	6.0E-5	B	2	NC	2.13E-7	T_5MQLD_1
59	46	2.9E-3	D	2	8.48E-4	2.46E-6*	T_5MLE
60	47	3.9E-9	-	2	NC	1.94E-10	T_5MLP
All Sequences						8.7578E-5	
Top 14 * Sequences only						8.3190E-5	
Next 46 Sequences						4.39E-6	

Table B.12 ANO-1 Results After Recovery - Top 23 Sequences

Ranking	Analysis Number	Core Melt Failure Sequence	Core Melt Probability	Boolean Sequence Number
1	8	$S_3MH_1H'_2$	2.24E-5	9
2	15	T_1MLE	2.13E-5	22
3	39	$T_3QH_1H'_2$	1.21E-5	1
4	10	S_3MED_2	6.60E-6	11
5	41	$T_3QD_1D_2$	4.57E-6	3a
6	51	T_4MLE	4.08E-6	38
7	59	T_5MLE	2.46E-6	46
8	17	T_2MLE	2.38E-6	30
9	11	S_3MXE	2.33E-6	12
10	27	$T_2MQH_1H'_2$	1.71E-6	1
11	19	T_3MLE	1.13E-6	30a
12	26	T_1MQLD_1	7.84E-7	21
13	33	$T_3MQH_1H'_2$	7.32E-7	1
14	29	$T_2MQD_1D_2$	6.66E-7	3a
15	24	T_1MQXD_1	5.94E-7	19
16	1	$S_2MH_1H'_2$	3.78E-7	1
17	35	$T_3MQD_1D_2$	2.76E-7	3a
18	42	T_3QXD_1	2.44E-7	4a
19	32	T_2MQLD_1	5.14E-8	7a
20	21	$T_1MQH_1H'_2$	3.66E-8	16
21	14	S_3MLE	3.50E-8	15
22	31	T_2MQLH_1	4.50E-10	5
23	12	S_3MLH_1	4.02E-10	13
23 Sequences Comparable to Table B.10			8.4806E-5	
Top 14 Sequences Only			8.3190E-5	

and summarized in Table B.12. The top 14 accident sequences account for more than 95% of the core melt probability after recovery factors have been applied. These accident sequences are the basis for the remaining analysis.

5.2.2 Quantification given Secondary Blowdown Not Applicable

The purpose of this section is to determine the effect of the secondary blowdown mode on core melt probability. To do this, first consider the way the accident sequences would appear without secondary blowdown and what sequences in the original quantification would be replaced by the new accident sequences. This is summarized in Table B.13. In each of the seven cases the new accident sequence replaces the three choices leading to core melt in the secondary blowdown branch by going directly to core melt, i.e., $\bar{L}D_1X$, $\bar{L}D_1\bar{X}D_2$, $\bar{L}D_1\bar{X}\bar{D}_2H_2$ all reduce to $\bar{L}D_1$. In one case the new accident sequence replaces only one.

The quantification for each of the accident sequences without secondary blowdown and before recovery is given in Table B.14 along with the sum of the three sequences it would replace. The net difference is $9.0E-5$. This analysis was not carried further into the recovery analysis, however, recovery previously gave an improvement of about 5.5. This factor would presumably be similar for the sequences with or without the inclusion of secondary blowdown.

5.2.3 Quantification Given Feed and Bleed Not Applicable

The purpose of this section is to determine the effect of the feed and bleed mode on core melt probability. This is accomplished by considering the way the accident sequences would appear without feed and bleed, and which of the original sequences would be replaced by the new accident sequences. This is summarized in Table B.15. In each of the first eight cases (i.e., analysis numbers 69 through 76) there are two or three choices leading to core melt in the feed and bleed branch, i.e., LD_1 , $\bar{L}\bar{D}_1P_1$ and $\bar{L}\bar{D}_1\bar{P}_1H_1$ replaced by L alone. In the last five cases which are solely transients, that is no induced LOCA, the Boolean sequence $L\bar{P}\bar{Q}E$ is replaced by $L\bar{P}\bar{Q}$. This is essentially bleed without feed and would eventually lead to core uncover.

The quantification for each of the accident sequences without feed and bleed and before recovery is given in Table B.16 along with the corresponding sum of the sequences it would replace. The net difference is $3.11E-3$ which demonstrates a significant improvement for having a feed and bleed capability. When recovery is applied to the accident sequences without feed and bleed, and to the dominant accident sequences with feed and bleed the net difference is $1.15E-3$. This is tabulated in Table B.17.

(Text continued on page B-76.)

Table B.13 ANO-1 Core Melt Accident Sequences
Without Secondary Blowdown

Anal. No.	Multi- plier	Boolean Events	ANO Seq. No.	Replaces Basecase Sequence Analysis Numbers
61	S_2^M	$\bar{L}D_1$	4	2, 3, and 4
62	S_3^M	$\bar{L}E$	12	9, 10, and 11
63	T_1^{MQ}	$\bar{L}D_1$	19	22
64	T_2^{MQ}	$\bar{L}D_1$	4a	28, 29, and 30
65	T_3^{MQ}	$\bar{L}D_1$	4a	34, 35, and 36
66	T_3^Q	$\bar{L}D_1$	4a	40, 41, and 42
67	T_4^{MQ}	$\bar{L}D_1$	35	46, 47, and 48
68	T_5^{MQ}	$\bar{L}D_1$	43	54, 55, and 56

Table B.14 ANO-1 Core Melt Accident Sequences
Probabilities Without Secondary Blowdown

Anal. No.	Multi- plier	Boolean Sequence	ANO Seq. No.	Multi. Freq.	Boolean Sequence Prob.	Core Melt Prob.	Sum of Core Melt Frequen- cies for Sequences Replaced
61	S ₂ M	$\bar{L}D_1$	4	3.1E-4	3.69E-3	1.21E-6	8.99E-7
62	S ₃ M	$\bar{L}E$	12	2.0E-2	6.89E-3	1.38E-4	5.95E-5
63	T ₁ MQ	$\bar{L}D_1$	19	1.2E-4	1.65E-2	1.98E-6	1.98E-6
64	T ₂ MQ	$\bar{L}D_1$	4a	1.4E-3	3.89E-3	5.45E-6	4.19E-6
65	T ₃ MQ	$\bar{L}D_1$	4a	6.0E-4	3.89E-3	2.33E-6	1.74E-6
66	T ₃ Q	$\bar{L}D_1$	4a	9.9E-3	3.89E-3	3.85E-5	2.88E-5
67	T ₄ MQ	$\bar{L}D_1$	35	8.4E-6	6.38E-2	5.36E-7	3.91E-7
68	T ₅ MQ	$\bar{L}D_1$	43	4.2E-6	6.33E-2	2.66E-7	1.95E-7
						-----	-----
						1.88E-4	9.77E-5

Table B.15 ANO-1 Core Melt Accident Sequences
Without Feed and Bleed

Anal. No.	Multi- plier	Boolean Events	ANO Seq. No.	Replaces Basecase Sequence Analysis Numbers	SWS Mod.	VB No.	ESF TSL	Seq RA	Rec Time
69	S ₂ M	L	8	5, 6, and 7	1	1	1	2	C
70	S ₃ M	L	8c	12, 13, and 14	1	1	1	2	E
71	T ₁ MPQ	L	56	25 and 26	1	2	1	1	B
72	T ₂ MPQ	L	8a	31 and 32	1	3	1	2	B
73	T ₃ MPQ	L	8b	37 and 38	1	3	1	-	B
74	T ₃ Q	L	8a	43 and 44	1	3	1	2	B
75	T ₄ MPQ	L	57	49 and 50	.	4	1	2	B
76	T ₅ MPQ	L	58	57 and 58	1	5	1	2	B
77	T ₁ MPQ̄	L	23	15	2	2	0	1	D
78	T ₂ MPQ̄	L	31	17	2	3	0	2	D
79	T ₃ MPQ̄	L	31a	19	2	3	0	-	D
80	T ₄ MPQ̄	L	39	51	2	4	0	2	D
81	T ₅ MPQ̄	L	47	59	2	5	0	2	D

Table B.16 AEO-1 Core Melt Accicent Sequence
Probabilities Without Feed and Bleed
(Before Recovery was Applied)

Anal. No.	Multi- plier	Boolean Sequence	ANO Seq. No.	Multi. Freq.	Boolean Sequence Prob.	Core Melt Prob.	Sum of Core Melt Prob. for Sequences Replaced
69	S ₂ M	L	8	3.1E-4	2.07E-3	6.41E-7	4.12E-8
70	S ₃ M	L	8c	2.0E-2	2.07E-3	4.14E-5	2.77E-6
71	T ₁ MPQ	L	56	1.7E-3	4.74E-3	8.05E-6	1.85E-6
72	T ₂ MPQ	L	8a	2.0E-2	2.07E-3	4.14E-5	2.66E-6
73	T ₃ MPQ	L	8b	8.5E-3	2.07E-3	1.76E-5	1.13E-6
74	T ₃ Q	L	8a	9.9E-3	2.07E-3	2.05E-5	1.31E-6
75	T ₄ MPQ	L	57	1.2E-4	3.67E-2	4.41E-6	4.98E-7
76	T ₅ MPQ	L	58	6.0E-5	4.99E-2	2.99E-6	3.59E-7
77	T ₁ MPQ̄	L	23	8.4E-2	4.31E-3	3.62E-4	7.27E-5
78	T ₂ MPQ̄	L	31	9.8E-1	1.74E-3	1.71E-3	5.65E-5
79	T ₃ MPQ̄	L	31a	4.2E-1	1.74E-3	7.33E-4	2.42E-5
80	T ₄ MPQ̄	L	39	5.9E-3	3.67E-2	2.17E-4	1.32E-5
81	T ₅ MPQ̄	L	47	2.9E-3	4.98E-2	1.44E-4	9.64E-6
						3.30E-3	1.87E-4

Table B.17 ANO-1 Core Melt Accident Sequence
Probabilities Without Feed and Bleed
(with recovery applied)

Anal. No.	Multi- plier	Boolean Sequence	ANO Seq. No.	Multi. Freq.	Boolean Sequence Prob.	Core Melt Prob.	Sum of Core Melt Prob. for Sequences Replaced
69	S_2M	L	8	3.1E-4	7.05E-4	2.19E-7	5.84E-10*
70	S_3M	L	8c	2.0E-2	4.17E-4	8.34E-6	3.54E-8
71	$T_1\bar{M}\bar{P}Q$	L	56	1.7E-3	2.19E-3	3.72E-6	8.11E-7**
72	$T_2\bar{M}\bar{P}Q$	L	8a	2.0E-2	7.29E-4	1.46E-5	5.19E-8
73	$T_3\bar{M}\bar{P}Q$	L	8b	8.5E-3	7.29E-4	6.20E-6	4.11E-8
74	T_3Q	L	8a	9.9E-3	7.29E-4	7.22E-6	2.56E-8
75	$T_4\bar{M}\bar{P}Q$	L	57	1.2E-4	3.63E-2	4.36E-6	2.84E-7*
76	$T_5\bar{M}\bar{P}Q$	L	58	6.0E-5	4.71E-2	2.83E-6	2.15E-7*
77	$T_1\bar{M}\bar{P}\bar{Q}$	L	23	8.4E-2	1.16E-3	9.74E-5	2.13E-5
78	$T_2\bar{M}\bar{P}\bar{Q}$	L	31	9.8E-1	5.14E-4	5.04E-4	2.38E-6
79	$T_3\bar{M}\bar{P}\bar{Q}$	L	31a	4.2E-1	5.71E-4	2.40E-4	1.13E-6
80	$T_4\bar{M}\bar{P}\bar{Q}$	L	39	5.9E-3	3.26E-2	1.92E-4	4.08E-6
81	$T_5\bar{M}\bar{P}\bar{Q}$	L	47	2.9E-3	3.57E-2	1.04E-4	2.46E-6
						1.18E-3	3.28E-5

* Recovery not applied to these accident sequences since their contribution to core melt was not significant. Therefore, the difference will be larger than that shown.

** One of two contributing sequences had recovery applied.

5.2.4 Quantification Given the PORVs are Unblocked

The basecase assumes the PORVs are blocked since there is no way to assure they are unblocked and our observation is that usually the ERV is blocked due to leakage. Thus the purpose of this sensitivity issue is to quantify this effect in terms of core melt probability. There are 20 accident sequences (82 through 101) affected by the ERV that also involve feed and bleed. These are described in Table B.18. There are also 5 accident sequences (102 through 106) affected by the ERV which do not involve feed and bleed as shown in Table B.19.

The quantification is given in Tables B.20 and B.21. There is no effect of ERV blocking for accident sequences involving feed and bleed (Table B.20). The values shown in Table B.21 indicate that the effect of unblocked PORVs is insignificant. Recovery was not applied and in our opinion would not change the results.

5.3 Long Term Station Blackout

Long term station blackout results from the failure of DC power due to battery depletion or failure of the EFWS water sources (i.e., the CST). Short term station blackout is covered in the basecase analysis by the random failures of support systems (electric power and service water in particular) and the EFWS turbine pump (TP) train. All of the components in the turbine pump train except the pump itself are effectively early failures. The turbine pump (defined as the pump, turbine, overpressure protection valve, and the governor-regulator valve) may fail early as the other components in the train or late due to battery or CST depletion.

Neither battery depletion nor CST depletion are a problem at ANO-1. The batteries are qualified for 8 hours. The CST capacity should last for over 8 hours in the worst case.

5.4 Vulnerability Identification

First, the cut sets were examined to identify the specific internal vulnerabilities, and then later the top 14 accident sequences were combined with the containment systems event tree and then later in the analysis extrapolated to measures of public risk.

The specific vulnerabilities were identified using Boolean event sequences derived from the SETS program. The top 14 accident sequences result from 11 Boolean event sequences. The raw output for these 11 Boolean event sequences showing the cut sets is given in Table B.22. Each of the top 14 accident sequences were examined for the most significant cut sets. Table B.23 tabulates these cut sets and their core melt frequencies with the residual core melt frequency for each

(Text continued on page B-110.)

Table B.18 ANO-1 Core Melt Accident Sequences with
Feed and Bleed with the ERV Unblocked

Anal. No.	Multiplier	Boolean Events	ANO Seq. No.	Replaced Basecase Seq. Analysis No.
82	S ₂ M	L $\bar{D}_1$ $\bar{P}_1$ H ₁	48	5
83	S ₂ M	L $\bar{D}_1$ P ₁	49	6
84	S ₃ M	L $\bar{E}$ $\bar{P}_1$ H ₁	50	12
85	S ₃ M	L $\bar{E}$ P ₁	51	13
86	T ₁ M $\bar{P}$ Q	L $\bar{D}_1$ $\bar{P}_1$ H ₁	52	25
87	T ₁ M $\bar{P}$ Q	LD ₁	21	26
88	T ₂ M $\bar{P}$ Q	L $\bar{D}_1$ $\bar{P}_1$ H ₁	48	31
89	T ₂ M $\bar{P}$ Q	LD ₁	7	32
90	T ₃ M $\bar{P}$ Q	L $\bar{D}_1$ $\bar{P}_1$ H ₁	48	37
91	T ₃ M $\bar{P}$ Q	LD ₁	7	38
92	T ₃ Q	L $\bar{D}_1$ $\bar{P}_1$ H ₁	48	43
93	T ₄ M $\bar{P}$ Q	L $\bar{D}_1$ $\bar{P}_1$ H ₁	54	49
94	T ₄ M $\bar{P}$ Q	LD ₁	37	50
95	T ₅ M $\bar{P}$ Q	L $\bar{D}_1$ $\bar{P}_1$ H ₁	55	57
96	T ₅ M $\bar{P}$ Q	LD ₁	45	58
97	T ₁ M $\bar{P}$ $\bar{Q}$	LE	22	15
98	T ₂ M $\bar{P}$ $\bar{Q}$	LE	30	17
99	T ₃ M $\bar{P}$ $\bar{Q}$	LE	30	19
100	T ₄ M $\bar{P}$ $\bar{Q}$	LE	38	51
101	T ₅ M $\bar{P}$ $\bar{Q}$	LE	46	59

Table B.19 ANO-1 Core Melt Accident Sequences other
than F&B Affected by Unblocked ERV

Anal. No.	Multiplier	Boolean Events	ANO Seq. No.	Replaced Basecase Seq. Analysis No.
102	T ₁ MP	L	23	16
103	T ₂ MP	L	31	18
104	T ₃ MP	L	31	20
105	T ₄ MP	L	39	52
106	T ₅ MP	L	47	60

Table B.20 ANO-1 Core Melt Accident Sequence Probabilities
For Feed and Bleed with the ERV Unblocked

Anal. No.	Multi- plier	Boolean Sequence	ANO Seq. No.	Multi. Prob.	Boolean Sequence Prob.	Core Melt Prob.	Core Melt Frequencies for Sequences Replaced
82	S_2M	$\bar{L}\bar{D}_1\bar{P}_1H_1$	48	$3.1E-4$	$5.74E-5$	$1.78E-8$	$1.78E-8$
83	S_2M	$\bar{L}\bar{D}_1P_1$	49	$3.1E-4$	ϕ	ϕ	ϕ
84	S_3M	$\bar{L}\bar{E}\bar{P}_1H_1$	50	$2.0E-2$	$5.74E-5$	$1.15E-6$	$1.15E-6$
85	S_3M	$\bar{L}\bar{E}\bar{P}_1$	51	$2.0E-2$	ϕ	ϕ	ϕ
86	$T_1\bar{M}\bar{P}Q$	$\bar{L}\bar{D}_1\bar{P}_1H_1$	52	$1.7E-3$	$1.56E-4$	$2.65E-7$	$2.65E-7$
87	$T_1\bar{M}\bar{P}Q$	$\bar{L}D_1$	21	$1.7E-3$	$9.29E-4$	$1.58E-6$	$1.58E-6$
88	$T_2\bar{M}\bar{P}Q$	$\bar{L}\bar{D}_1\bar{P}_1H_1$	48	$2.0E-2$	$5.74E-5$	$1.15E-6$	$1.15E-6$
89	$T_2\bar{M}\bar{P}Q$	$\bar{L}D_1$	7	$2.0E-2$	$7.54E-5$	$1.51E-6$	$1.51E-6$
90	$T_3\bar{M}\bar{P}Q$	$\bar{L}\bar{D}_1\bar{P}_1H_1$	48	$8.5E-3$	$5.74E-5$	$4.88E-7$	$4.88E-7$
91	$T_3\bar{M}\bar{P}Q$	$\bar{L}D_1$	7	$8.5E-3$	$7.54E-5$	$6.41E-7$	$6.41E-7$
92	T_3Q	$\bar{L}\bar{D}_1\bar{P}_1H_1$	48	$9.9E-3$	$5.74E-5$	$5.68E-7$	$5.68E-7$
93	$T_4\bar{M}\bar{P}Q$	$\bar{L}\bar{D}_1\bar{P}_1H_1$	54	$1.2E-4$	$1.80E-3$	$2.16E-7$	$2.16E-7$
94	$T_4\bar{M}\bar{P}Q$	$\bar{L}D_1$	37	$1.2E-4$	$2.35E-3$	$2.82E-7$	$2.82E-7$
95	$T_5\bar{M}\bar{P}Q$	$\bar{L}\bar{D}_1\bar{P}_1H_1$	55	$6.0E-5$	$2.43E-3$	$1.46E-7$	$1.46E-7$
96	$T_5\bar{M}\bar{P}Q$	$\bar{L}D_1$	45	$6.0E-5$	$3.55E-3$	$2.13E-7$	$2.13E-7$
97	$T_1\bar{M}\bar{P}\bar{Q}$	LE	22	$8.4E-2$	$8.65E-4$	$7.27E-5$	$7.27E-5$
98	$T_1\bar{M}\bar{P}\bar{Q}$	LE	30	$9.8E-1$	$5.77E-5$	$5.65E-5$	$5.65E-5$
99	$T_2\bar{M}\bar{P}\bar{Q}$	LE	30	$4.2E-1$	$5.77E-5$	$2.42E-5$	$2.42E-5$
100	$T_2\bar{M}\bar{P}\bar{Q}$	LE	38	$5.9E-3$	$2.23E-3$	$1.32E-5$	$1.32E-5$
101	$T_5\bar{M}\bar{P}\bar{Q}$	LE	46	$2.9E-3$	$3.32E-3$	$9.64E-6$	$9.64E-6$
						$1.84E-5$	$1.84E-5$

Table B.21 ANO-1 Core Melt Accident Sequences Other Than
Feed and Bleed Affected by Unblocked ERV

Anal. No.	Multi- plier	Boolean Sequence	ANO Seq. No.	Multi.* Freq.	Boolean Sequence Freq.	Core Melt Freq.	Core Melt Frequencies for Sequences Replaced
102	T ₁ MP	L	23	3.3E-11	4.31E-3	c	4.74E-10
103	T ₂ MP	L	31	3.9E-10	1.74E-3	c	2.26E-9
104	T ₃ MP	L	31	1.7E-10	1.74E-3	c	9.57E-10
105	T ₄ MP	L	39	2.3E-12	3.67E-2	c	2.86E-10
106	T ₅ MP	L	47	1.2E-12	4.98E-2	c	1.94E-10

* Each multiplier probability is decreased from the previous value by 3E-4 since the ERV is available to relieve pressure.

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Table B.22 ANO-1 Boolean Accident Sequences
After Recovery Has Been Applied

ANON1 =

1	1.5200E-04	PA-VCCH * SWSV-40-41-CM +
2	1.5200E-04	RA-VCCH * LPRSV-CM +
3	1.5200E-04	SWSV-02-03-CM * RA-VCCH +
4	7.5690E-05	LPIOP34A-PMD-LF * LPIOP34B-PMD-LF +
5	7.0000E-05	LPSP-CM +
6	2.6448E-05	RA-VCCH * LPIOP34A-PMD-LF * SWS3802-VCC-LF +
7	2.6448E-05	RA-VCCH * LPIOP34A-PMD-LF * SWS3841-NCC-LF +
8	2.6448E-05	RA-VCCH * LPIOP34B-PMD-LF * LPR1405A-VCC-LF +
9	2.6448E-05	RA-VCCH * LPIOP34B-PMD-LF * SWS3840-NCC-LF +
10	2.6448E-05	RA-VCCH * LPIOP34B-PMD-LF * LPI1407A-VCC-LF +
11	2.6448E-05	RA-VCCH * LPIOP34A-PMD-LF * LPR1406B-VCC-LF +
12	2.6448E-05	RA-VCCH * LPIOP34B-PMD-LF * SWS3803-VCC-LF +
13	2.6448E-05	RA-VCCH * LPIOP34A-PMD-LF * LPI1408B-VCC-LF +
14	9.2416E-06	RA-VCCH * LPR1406B-VCC-LF * LPI1407A-VCC-LF * RA-VCCH2 +
15	9.2416E-06	RA-VCCH * LPR1406B-VCC-LF * SWS3840-NCC-LF * RA-VCCH2 +
16	9.2416E-06	RA-VCCH * SWS3802-VCC-LF * LPR1405A-VCC-LF * RA-VCCH2 +
17	9.2416E-06	RA-VCCH * SWS3803-VCC-LF * SWS3802-VCC-LF * RA-VCCH2 +
18	9.2416E-06	RA-VCCH * SWS3841-NCC-LF * LPI1407A-VCC-LF * RA-VCCH2 +
19	9.2416E-06	RA-VCCH * SWS3841-NCC-LF * SWS3840-NCC-LF * RA-VCCH2 +
20	9.2416E-06	RA-VCCH * LPR1405A-VCC-LF * LPI1408B-VCC-LF * RA-VCCH2 +

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Table B.22 ANO-1 Boolean Accident Sequences After
Recovery Has Been Applied (continued)

21	9.2416E-06	RA-VCCH * SWS3803-VCC-LF * LPI1408B-VCC-LF * RA-VCCH2 +
22	9.2416E-06	RA-VCCH * LPR1406B-VCC-LF * LPR1405A-VCC-LF * RA-VCCH2 +
23	9.2416E-06	RA-VCCH * SWS3803-VCC-LF * LPR1406B-VCC-LF * RA-VCCH2 +
24	9.2416E-06	RA-VCCH * SWS3802-VCC-LF * LPI1407A-VCC-LF * RA-VCCH2 +
25	9.2416E-06	RA-VCCH * SWS3802-VCC-LF * SWS3840-NCC-LF * RA-VCCH2 +
26	9.2416E-06	RA-VCCH * SWS3841-NCC-LF * LPR1405A-VCC-LF * RA-VCCH2 +
27	9.2416E-06	RA-VCCH * SWS3803-VCC-LF * SWS3841-NCC-LF * RA-VCCH2 +
28	9.2416E-06	RA-VCCH * SWS3840-NCC-LF * LPI1408B-VCC-LF * RA-VCCH2 +
29	9.0000E-06	HPRS-MANACT * LPRS-MANACT +
30	2.4360E-06	LPIOP34A-PMD-LF * LPIOP34B-PMD-UTH +
31	2.4360E-06	LPIOP34B-PMD-LF * LPIOP34A-PMD-UTH +
32	1.6530E-06	LF-SWS-VCH4A * LPIOP34A-PMD-LF * RA-7H +
722	3.0400E-09	RA-VCCH * RA-7H * LPR1405A-VCC-LF * LF-ECS-A2 +
723	3.0400E-09	RA-VCCH * RA-7H * SWS3803-VCC-LF * LF-ECS-A2 +
724	3.0400E-09	RA-VCCH * RA-7H * SWS3802-VCC-LF * LF-ECS-B4 +
725	3.1400E-09	RA-VCCH * RA-7H * SWS3840-NCC-LF * LF-ECS-A4 +
726	3.0400E-09	RA-VCCH * RA-7H * SWS3841-NCC-LF * LF-ECS-B2

THE MAXIMUM TERM VALUE FOR COMPUTATION 1 IS 1.520000000000E-04
(THE SUM OF THE TERM VALUES IS 1.218733327518E-03)

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TERM
NUMBER

PROB.
OF TERM

Table B.22 ANO-1 Boolean Accident Sequences After
Recovery Has Been Applied (continued)

ANOR3A =

1	2.2400E-04	HPISV-CM * RA-VCCB +
2	1.0000E-04	LPIBWIX-XOC-LF +
3	3.5840E-05	LPI1407A-VCC-LF * LPI1408B-VCC-LF * RA-VCCB +
4	1.0000E-05	SWSP-CM +
5	4.5600E-06	LPI1408B-VCC-LF * SWSOP4BA-PMD-LF * RA-6B +
6	4.5600E-06	LPI1407A-VCC-LF * SWSOP4CB-PMD-LF * RA-6B +
7	4.5600E-06	LPI1407A-VCC-LF * SWSOP4CB-PAC-LF * RA-6B +
8	4.5600E-06	LPI1408B-VCC-LF * SWSOP4BA-PAC-LF * RA-6B +
9	3.6100E-06	LF-SWS-VCH4A * LF-SWS-VCH4B * RA-7G +
10	3.6000E-06	SWS3824-VOC-LF * RA-6B +
11	3.2490E-06	SWSOP4BA-PMD-LF * SWSOP4CB-PMD-LF * RA-6B +
12	3.2490E-06	SWSOP4BA-PMD-LF * SWSOP4CB-PAC-LF * RA-6B +
13	3.2490E-06	SWSOP4CB-PMD-LF * SWSOP4BA-PAC-LF * RA-6B +
14	3.2490E-06	SWSOP4BA-PAC-LF * SWSOP4CB-PAC-LF * RA-6B +
15	2.4000E-06	LPI1408B-VCC-LF * SWS3643A-V00-LF * RA-6B +
16	1.7100E-06	SWSOP4CB-PMD-LF * SWS3643A-V00-LF * RA-6B +
17	1.7100E-06	SWSOP4CB-PAC-LF * SWS3820A-V00-LF * RA-6B +
18	1.7100E-06	SWSOP4CB-PAC-LF * SWS3643A-V00-LF * RA-6B +
19	1.5200E-06	LF-SWS-VCH4A * LPI1407A-VCC-LF * RA-7G +
20	1.5200E-06	LF-SWS-VCH4B * LPI1408B-VCC-LF * RA-7G +

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Table B.22 ANO-1 Boolean Accident Sequences After
Recovery Has Been Applied (continued)

21	1.2160E-06	LF-SWS-VCH4A * LF-ECS-B6 * RA-7G +
22	1.0830E-06	LF-SWS-VCH4B * SWSOP4CB-PMD-LF * RA-7G +
23	1.0830E-06	LF-SWS-VCH4A * SWSOP4BA-PMD-LF * RA-7G +
24	1.0830E-06	LF-SWS-VCH4B * SWSOP4CB-PAC-LF * RA-7G +
25	1.0830E-06	LF-SWS-VCH4A * SWSOP4BA-PAC-LF * RA-7G +
26	1.0500E-06	LOSS-OSP * DG-CM * RA-1B +
27	1.0108E-06	LOSS-OSP * DG1-GEN-LF * DG2-GEN-LF * RA-1B +
28	9.1200E-07	LF-SWS-VCH4B * LF-ECS-A6 * RA-7G +
29	5.7000E-07	LF-SWS-VCH4A * SWS3643A-V00-LF * RA-7G +
30	5.7000E-07	LF-SWS-VCH4A * RA-7G * SWS3820A-V00-LF +
31	5.1200E-07	LPI1408B-VCC-LF * LF-ECS-B6 * RA-7G +
32	4.9290E-07	LPI1407A-VCC-LF * SWS002CX-XOC-UTM * RA-VCCB +
317	4.8000E-09	LF-ECS-A6 * LF-ECS-B4 * RA-7G +
318	4.4544E-09	LPIOP34B-PMD-LF * LF-ECS-B6 * RA-7G * SWS3810B-VCC-LF +
319	4.4544E-09	SWS3841-NCC-LF * LF-ECS-B6 * RA-7G * HPIP36CB-PMD-LF +
320	4.4544E-09	LF-ECS-B6 * LPI1400B-VCC-LF * RA-7G * HPIP36CB-PMD-LF +
321	4.0960E-09	SWS3841-NCC-LF * LF-ECS-B6 * RA-7G * SWS3810B-VCC-LF

THE MAXIMUM TERM VALUE FOR COMPUTATION 1 IS 2.240000000000E-04
(THE SUM OF THE TERM VALUES IS 4.760017290598E-04)

TERM
NUMBER

PROB.
OF TERM

Table B.22 ANO-1 Boolean Accident Sequences After
Recovery Has Been Applied (continued)

ANOR9 =

1	1.5200E-04	RA-VCCH * SWSV-40-41-CM *
2	1.5200E-04	RA-VCCH * LPRSV-CM *
3	1.5200E-04	SWSV-02-03-CM * RA-VCCH *
4	7.3690E-05	LPIOP34A-PMD-LF * LPIOP34B-PMD-LF *
5	7.0000E-05	LPSP-CM *
6	2.6448E-05	RA-VCCH * LPIOP34A-PMD-LF * SWS3802-VCC-LF *
7	2.6448E-05	RA-VCCH * LPIOP34B-PMD-LF * SWS3803-VCC-LF *
8	2.6448E-05	RA-VCCH * LPIOP34B-PMD-LF * SWS3840-NCC-LF *
9	2.6448E-05	RA-VCCH * LPIOP34A-PMD-LF * SWS3841-NCC-LF *
10	2.6448E-05	RA-VCCH * LPIOP34B-PMD-LF * LPI1407A-VCC-LF *
11	2.6448E-05	RA-VCCH * LPIOP34A-PMD-LF * LPI1408B-VCC-LF *
12	2.6448E-05	RA-VCCH * LPIOP34A-PMD-LF * LPR1406B-VCC-LF *
13	2.6448E-05	RA-VCCH * LPIOP34B-PMD-LF * LPR1405A-VCC-LF *
14	9.2416E-06	RA-VCCH * LPR1406B-VCC-LF * LPI1407A-VCC-LF * RA-VCCH2 *
15	9.2416E-06	RA-VCCH * LPR1406B-VCC-LF * SWS3840-NCC-LF * RA-VCCH2 *
16	9.2416E-06	RA-VCCH * SWS3802-VCC-LF * LPR1405A-VCC-LF * RA-VCCH2 *
17	9.2416E-06	RA-VCCH * SWS3803-VCC-LF * SWS3802-VCC-LF * RA-VCCH2 *
18	9.2416E-06	RA-VCCH * SWS3841-NCC-LF * LPI1407A-VCC-LF * RA-VCCH2 *
19	9.2416E-06	RA-VCCH * SWS3841-NCC-LF * SWS3840-NCC-LF * RA-VCCH2 *
20	9.2416E-06	RA-VCCH * LPR1406B-VCC-LF * LPR1405A-VCC-LF * RA-VCCH2 *

Table B.22 ANO-1 Boolean Accident Sequences After
Recovery Has Been Applied (continued)

21	9.2416E-06	RA-VCCH * SWS3803-VCC-LF * LPR1406B-VCC-LF * RA-VCCH2 *
22	9.2416E-06	RA-VCCH * SWS3802-VCC-LF * LPI1407A-VCC-LF * RA-VCCH2 *
23	9.2416E-06	PA-VCCH * SWS3802-VCC-LF * SWS3840-NCC-LF * RA-VCCH2 *
24	9.2416E-06	RA-VCCH * LPR1405A-VCC-LF * LPI1408B-VCC-LF * RA-VCCH2 *
25	9.2416E-06	PA-VCCH * SWS3803-VCC-LF * LPI1408B-VCC-LF * RA-VCCH2 *
26	9.2416E-06	RA-VCCH * SWS3840-NCC-LF * LPI1408B-VCC-LF * RA-VCCH2 *
27	9.2416E-06	RA-VCCH * SWS3841-NCC-LF * LPR1405A-VCC-LF * RA-VCCH2 *
28	9.2416E-06	RA-VCCH * SWS3803-VCC-LF * SWS3841-NCC-LF * RA-VCCH2 *
29	9.0000E-06	HPRS-MANACT * LPRS-MANACT *
30	2.4360E-06	LPIOP34A-PMD-LF * LPIOP34B-PMD-UTM *
31	2.4360E-06	LPIOP34B-PMD-LF * LPIOP34A-PMD-UTM *
32	1.6530E-06	LF-SWS-VCH4A * LPIOP34A-PMD-LF * RA-7H *
726	1.0944E-09	RA-VCCH * SWS3803-VCC-LF * RA-6H * SWS001CX-COC-LF *
727	1.0944E-09	RA-VCCH * SWS3802-VCC-LF * RA-6H * SWS001BX-COC-LF *
728	1.0944E-09	RA-VCCH * SWS3841-NCC-LF * RA-6H * SWS001BX-COC-LF *
729	3.2300E-11	RA-7H * LF-SWS-VCH4B * RA-6H * LF-ESF-T4 *
730	3.2300E-11	LF-SWS-VCH4A * RA-7H * RA-6H * LF-ESF-T3

THE MAXIMUM TERM VALUE FOR COMPUTATION 1 IS 1.520000000000E-04
(THE SUM OF THE TERM VALUES IS 1.119584325199E-03)

TERM
NUMBER

PROB.
OF TERM

Table B.22 ANO-1 Boolean Accident Sequences After
Recovery Has Been Applied (continued)

ANOR11 =

1	1.5600E-04	HPISV-CM * RA-VCCE +
2	1.0000E-04	LPIRW1X-XOC-LF +
3	2.4960E-05	LPI1407A-VCC-LF * LPI1408B-VCC-LF * RA-VCCE +
4	1.0000E-05	SWSP-CM +
5	3.6100E-06	LF-SWS-VCH4A * LF-SWS-VCH4B * RA-7G +
6	1.5200E-06	LF-SWS-VCH4B * LPI1408B-VCC-LF * RA-7G +
7	1.5200E-06	LF-SWS-VCH4A * LPI1407A-VCC-LF * RA-7G +
8	1.3680E-06	LPI1408B-VCC-LF * SWSOP4BA-PMD-LF * RA-6E +
9	1.3680E-06	LPI1408B-VCC-LF * SWSOP4BA-PAC-LF * RA-6E +
10	1.3680E-06	LPI1407A-VCC-LF * SWSOP4CB-PMD-LF * RA-6E +
11	1.3680E-06	LPI1407A-VCC-LF * SWSOP4CB-PAC-LF * RA-6E +
12	1.2160E-06	LF-SWS-VCH4A * LF-ECS-B6 * RA-7G +
13	1.0830E-06	LF-SWS-VCH4B * SWSOP4CB-PAC-LF * RA-7G +
14	1.0830E-06	LF-SWS-VCH4A * SWSOP4BA-PMD-LF * RA-7G +
15	1.0830E-06	LF-SWS-VCH4B * SWSOP4CB-PMD-LF * RA-7G +
16	1.0830E-06	LF-SWS-VCH4A * SWSOP4BA-PAC-LF * RA-7G +
17	1.0800E-06	SWS3824-VOC-LF * RA-6E +
18	9.7470E-07	SWSOP4BA-PMD-LF * SWSOP4CB-PAC-LF * RA-6E +
19	9.7470E-07	SWSOP4BA-PAC-LF * SWSOP4CB-PAC-LF * RA-6E +
20	9.7470E-07	SWSOP4BA-PMD-LF * SWSOP4CB-PMD-LF * RA-6E +

Table B.22 ANO-1 Boolean Accident Sequences After
Recovery Has Been Applied (continued)

21	9.7470E-07	SWSOP4CB-PMD-LF * SWSOP4BA-PAC-LF * RA-6E *
22	9.1200E-07	LF-SWS-VCH4B * LF-ECS-A6 * RA-7G *
23	7.2000E-07	LPI1408B-VCC-LF * SWS3643A-V00-LF * RA-6E *
24	5.7000E-07	LF-SWS-VCH4A * SWS3643A-V00-LF * RA-7G *
25	5.7000E-07	LF-SWS-VCH4A * RA-7G * SWS3820A-V00-LF *
26	5.1300E-07	SWSOP4CB-PMD-LF * SWS3643A-V00-LF * RA-6E *
27	5.1300E-07	SWSOP4CB-PAC-LF * SWS3820A-V00-LF * RA-6E *
28	5.1300E-07	SWSOP4CB-PAC-LF * SWS3643A-V00-LF * RA-6E *
29	5.1200E-07	LPI1408B-VCC-LF * LF-ECS-B6 * RA-7G *
30	4.6900E-07	SWSV-40-41-CM * RA-VCCE * FNB-MANACT *
31	4.0500E-07	LOSS-OSP * DG-CM * RA-10E * RA-1E *
32	3.8988E-07	LOSS-OSP * DG1-GEN-LF * DG2-GEN-LF * RA-1E * RA-11E *
354	1.9494E-09	SWSOP4BA-PAC-LF * LOSS-OSP * DG2-GEN-LF * RA-6E * RA-1E *
355	1.9494E-09	SWSOP4CB-PAC-LF * LOSS-OSP * DG1-GEN-LF * RA-6E * RA-1E *
356	1.9494E-09	SWSOP4CB-PMD-LF * LOSS-OSP * DG1-GEN-LF * RA-6E * RA-1E *
357	1.9494E-09	SWSOP4BA-PMD-LF * LOSS-OSP * DG2-GEN-LF * RA-6E * RA-1E *
358	7.2960E-10	LF-ECS-B6 * LOSS-OSP * DG2-GEN-LF * RA-7G * RA-1E

THE MAXIMUM TERM VALUE FOR COMPUTATION 1 IS 1.560000000000E-04
(THE SUM OF THE TERM VALUES IS 3.472478291397E-04)

TERM
NUMBER

PROB.
OF TERM

Table B.22 ANO-1 Boolean Accident Sequences After
Recovery Has Been Applied (continued)

ANOR12 =

1	3.0000E-03	FNB-MANACT +
2	1.5500E-04	HPISV-CM * RA-VCCE +
3	1.0000E-04	LPIBW1X-XCC-LF +
4	3.9000E-05	HPSV-CM * RA-VCCE +
5	3.0000E-05	HPSP-CM +
6	2.7144E-05	LPI1407A-VCC-LF * HPIP36CB-PMD-LF * RA-VCCE +
7	2.4960E-05	LPI1407A-VCC-LF * SWS3810B-VCC-LF * RA-VCCE +
8	2.4960E-05	LPI1407A-VCC-LF * LPI1408B-VCC-LF * RA-VCCE +
9	3.9000E-06	SWSP-CM * RA-VCCE +
10	3.6100E-06	LF-SWS-VCH4A * LF-SWS-VCH4B * RA-7G +
11	1.6530E-06	LF-SWS-VCH4B * RA-7G * HPIP36CB-PMD-LF +
12	1.5200E-06	LF-SWS-VCH4A * LPI1407A-VCC-LF * RA-7G +
13	1.5200E-06	LF-SWS-VCH4B * RA-7G * SWS3810B-VCC-LF +
14	1.5200E-06	LF-SWS-VCH4B * LPI1408B-VCC-LF * RA-7G +
15	1.4877E-06	SWSOP4BA-PAC-LF * HPIP36CB-PMD-LF * RA-6E +
16	1.4877E-06	SWSOP4BA-PMD-LF * HPIP36CB-PMD-LF * RA-6E +
17	1.3680E-06	SWSOP4BA-PMD-LF * SWS3810B-VCC-LF * RA-6E +
18	1.3680E-06	LPI1408B-VCC-LF * SWSOP4BA-PMD-LF * RA-6E +
19	1.3680E-06	SWSOP4BA-PAC-LF * SWS3810B-VCC-LF * RA-6E +
20	1.3680E-06	LPI1408B-VCC-LF * SWSOP4BA-PAC-LF * RA-6E +

Table B.22 ANO-1 Boolean Accident Sequences After
Recovery Has Been Applied (continued)

21	1.3680E-06	LPI1407A-VCC-LF * SWSOP4CB-PMD-LF * RA-6E +
22	1.3680E-06	LPI1407A-VCC-LF * SWSOP4CB-PAC-LF * RA-6E +
23	1.3572E-06	SWSV-08-10-CM * HPIP36BA-PMD-LF * RA-VCCE +
24	1.2480E-06	LPI1407A-VCC-LF * SWSV-08-10-CM * RA-VCCE +
25	1.2160E-06	LF-SWS-VCH4A * LF-ECS-B6 * RA-7G +
26	1.0830E-06	LF-SWS-VCH4A * SWSOP4BA-PAC-LF * RA-7G +
27	1.0830E-06	LF-SWS-VCH4B * SWSOP4CB-PMD-LF * RA-7G +
28	1.0830E-06	LF-SWS-VCH4A * SWSOP4BA-PMD-LF * RA-7G +
29	1.0830E-06	LF-SWS-VCH4B * SWSOP4CB-PAC-LF * RA-7G +
30	1.0800E-06	SWS3824-VOC-LF * RA-6E +
31	9.7470E-07	SWSOP4BA-PMD-LF * SWSOP4CB-PMD-LF * RA-6E +
32	9.7470E-07	SWSOP4CB-PMD-LF * SWSOP4BA-PAC-LF * RA-6E +
449	5.2900E-09	LF-ECS-A6 * SWS3645A-VOC-UTM * RA-7G +
450	5.2800E-09	LF-ECS-A6 * SWS3644A-VOC-UTM * RA-7G +
451	5.2800E-09	LF-ECS-A6 * RA-7G * HPI0BW3X-XOC-UTM +
452	2.1660E-09	LF-SWS-VCH4B * LOSS-OSP * DG2-GEN-LF * RA-7G * RA-1E +
453	2.1660E-09	LF-SWS-VCH4A * LOSS-OSP * DG1-GEN-LF * RA-7G * RA-1E

THE MAXIMUM TERM VALUE FOR COMPUTATION 1 IS 3.000000000000E-03
(THE SUM OF THE TERM VALUES IS 3.524769968497E-03)

TERM
NUMBER

PROB.
OF TERM

Table B.22 ANO-1 Boolean Accident Sequences After
Recovery Has Been Applied (continued)

ANOR21 =

1	3.0800E-04	BATT-CM * RA-13 * RA-8B *
2	3.6750E-05	RA-1B * DG-CM * EFWOP7AX-PTD-LF * RA-10B *
3	3.3378E-05	RA-1B * EFWOP7AX-PTD-LF * DG1-GEN-LF * DG2-GEN-LF * RA-11B *
4	8.0997E-06	RA-1B * EFWOP7AX-PTD-LF * DG1-GEN-LF * RA-11B * HP1P36CB-PMD-LF *
5	5.6791E-06	RA-1B * EFWOP7AX-PTD-LF * DG1-GEN-LF * RA-11B * DG2-GEN-UTM *
6	5.6791E-06	RA-1B * EFWOP7AX-PTD-LF * DG2-GEN-LF * RA-11B * DG1-GEN-UTM *
7	5.4880E-06	RA-1B * EFWOP7AX-PTD-LF * SMSV-06-07-CM * RA-VCCB *
8	4.1709E-06	RA-1B * EFWOP7AX-PTD-LF * DG2-GEN-LF * RA-VCCB * SMS3806A-VCC-LF *
9	4.1709E-06	RA-1B * EFWOP7AX-PTD-LF * DG1-GEN-LF * RA-VCCB * LPI1408B-VCC-LF *
10	4.1709E-06	RA-1B * EFWOP7AX-PTD-LF * DG1-GEN-LF * RA-VCCB * SMS3807B-VCC-LF *
11	4.1709E-06	RA-1B * EFWOP7AX-PTD-LF * DG1-GEN-LF * RA-VCCB * SMS3810B-VCC-LF *
12	2.2680E-06	RA-1B * BATTD07-ZBT-LF * BATTD06-ZBT-LF * RA-9B *
13	1.6758E-06	RA-1B * EFWOP7AX-PTD-LF * DG2-GEN-LF * RA-11B * BATTD07-ZBT-LF *
14	1.6758E-06	RA-1B * EFWOP7AX-PTD-LF * DG1-GEN-LF * RA-11B * BATTD06-ZBT-LF *
15	1.3860E-06	RA-1B * BATTD07-ZBT-LF * RA-9B * BATTD06-ZBT-UTM *
16	1.3860E-06	RA-1B * BATTD06-ZBT-LF * RA-9B * BATTD07-ZBT-UTM *
17	1.3002E-06	RA-1B * EFWOP7AX-PTD-LF * HP1P36CB-PMD-LF * DG1-GEN-UTM *
18	1.0241E-06	RA-1B * EFWOP7AX-PTD-LF * DG1-GEN-LF * RA-11B * BATTD06-ZBT-UTM *
19	1.0241E-06	RA-1B * EFWOP7AX-PTD-LF * DG2-GEN-LF * RA-11B * BATTD07-ZBT-UTM *
20	9.5491E-07	RA-1B * EFWOP7AX-PTD-LF * RA-VCCB * HP1P36CB-PMD-LF * SMS3806A-VCC-LF *

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Table B.22 ANO-1 Boolean Accident Sequences After
Recovery Has Been Applied (continued)

21	8.7809E-07	RA-1B * EFWOP7AX-PTD-LF * RA-VCCB * SWS3806A-VCC-LF * LPI1408B-VCC-LF *
22	8.7809E-07	RA-1B * EFWOP7AX-PTD-LF * RA-VCCB * SWS3806A-VCC-LF * SWS3807B-VCC-LF *
23	8.7808E-07	RA-1B * EFWOP7AX-PTD-LF * RA-VCCB * SWS3806A-VCC-LF * SWS3810B-VCC-LF *
24	8.0997E-07	RA-1B * EFWOP7AX-PTD-LF * DG1-GEN-LF * RA-6B * SWSOF4CB-FMC-LF *
25	8.0997E-07	RA-1B * EFWOP7AX-PTD-LF * DG1-GEN-LF * RA-6B * SWSOF4CB-FAC-LF *
26	8.0997E-07	RA-1B * EFWOP7AX-PTD-LF * DG2-GEN-LF * SWSOP4BA-FAC-LF * RA-6B *
27	7.1400E-07	RA-1B * DG-CM * RA-10B * EFWOP7AX-PTD-UTM *
28	6.8734E-07	RA-1B * DG1-GEN-LF * DG2-GEN-LF * RA-11B * EFWOP7AX-PTD-UTM *
29	6.6954E-07	RA-1B * EFWOP7AX-PTD-LF * RA-VCC3 * LPI1408B-VCC-LF * DG1-GEN-UTM *
30	6.6954E-07	RA-1B * EFWOP7AX-PTD-LF * RA-VCC3 * SWS3807B-VCC-LF * DG1-GEN-UTM *
31	6.6954E-07	RA-1B * EFWOP7AX-PTD-LF * RA-VCCB * SWS3810B-VCC-LF * DG1-GEN-UTM *
32	6.6954E-07	RA-1B * EFWOP7AX-PTD-LF * RA-VCC3 * SWS3806A-VCC-LF * DG2-GEN-UTM *
...		
219	1.8620E-09	RA-1B * EFWOP7AX-PTD-LF * LF-SWS-VCH4B * RA-7G * SWSV-40-42-CM *
220	1.3620E-09	RA-1B * EFWOP7AX-PTD-LF * LF-SWS-VCH4B * RA-7G * SWSV-09-10-CM *
221	1.8620E-09	RA-1B * EFWOP7AX-PTD-LF * LF-SWS-VCH4B * RA-7G * HPISV-CM *
222	1.7248E-09	RA-1B * EFWOP7AX-PTD-LF * RA-7G * LF-ECS-B6 * BATD06-ZRT-UTM *
223	1.7184E-09	RA-1B * LF-SWS-VCH4B * RA-7G * LF-SWS-VCH4A * EFWOP7AX-PTD-UTM *

THE MAXIMUM TERM VALUE FOR COMPUTATION 1 IS 3.080000000000E-04 .
(THE SUM OF THE TERM VALUES IS 4.614834335298E-04)

TERM
NUMBER

PROR.
OF TERM

Table B.22 ANO-1 Boolean Accident Sequences After
Recovery Has Been Applied (continued)

ANOR22 =

1	1.7600E-04	BATT-CM * RA-1D * RA-8D +
2	2.1000E-05	RA-1D * DG-CM * EFWOP7AX-PTD-LF * RA-10D +
3	2.0216E-05	RA-1D * EFWOP7AX-PTD-LF * DG1-GEN-LF * DG2-GEN-LF * RA-11D +
4	4.6284E-06	RA-1D * EFWOP7AX-PTD-LF * DG1-GEN-LF * RA-11D * HPIP36CB-PMD-LF +
5	3.2452E-06	PA-1D * EFWOP7AX-PTD-LF * DG2-GEN-LF * RA-11D * DG1-GEN-UTM +
6	3.2452E-06	RA-1D * EFWOP7AX-PTD-LF * DG1-GEN-LF * RA-11D * DG2-GEN-UTM +
7	2.2960E-06	RA-1D * EFWOP7AX-PTD-LF * SWSV-06-07-CM * RA-VCCD +
8	1.7450E-06	RA-1D * EFWOP7AX-PTD-LF * DG1-GEN-LF * RA-VCCD * LPI1408B-VCC-LF +
9	1.7450E-06	RA-1D * EFWOP7AX-PTD-LF * DG1-GEN-LF * RA-VCCD * SWS3807B-VCC-LF +
10	1.5960E-06	RA-1D * EFWOP7AX-PTD-LF * DG1-GEN-LF * RA-11D * FNB-MANACT +
11	1.0368E-06	RA-1D * BATTD07-ZBT-LF * BATTD06-ZBT-LF * RA-9D +
12	7.6608E-07	RA-1D * EFWOP7AX-PTD-LF * DG2-GEN-LF * BATTD07-ZBT-LF * RA-9D +
13	7.6608E-07	RA-1D * EFWOP7AX-PTD-LF * DG1-GEN-LF * BATTD06-ZBT-LF * RA-9D +
14	7.4298E-07	RA-1D * EFWOP7AX-PTD-LF * HPIP36CB-PMD-LF * DG1-GEN-UTM +
15	6.3360E-07	RA-1D * BATTD06-ZBT-LF * RA-9D * BATTD07-ZBT-UTM +
16	6.3360E-07	RA-1D * BATTD07-ZBT-LF * RA-9D * BATTD06-ZBT-UTM +
17	5.8520E-07	RA-1D * EFWOP7AX-PTD-LF * DG2-GEN-LF * RA-11D * BATTD07-ZBT-UTM +
18	5.8520E-07	RA-1D * EFWOP7AX-PTD-LF * DG1-GEN-LF * RA-11D * BATTD06-ZBT-UTM +
19	5.0470E-07	RA-1D * EFWOP7AX-PTD-LF * SWS3824-VOC-LF +
20	4.7400E-07	RA-1D * DG-CM * RA-10D * EFWOP7AX-PTD-UTM +

Table B.22 ANO-1 Boolean Accident Sequences After Recovery Has Been Applied (continued)

21	3.9950E-07	RA-1D * EFWOP7AX-PTD-LF * RA-VCCD * HPIP36CB-PMO-LF * SWS3806A-VCC-LF *
22	3.9277E-07	RA-1D * DG1-GEN-LF * DG2-GEN-LF * RA-11D * EFWOP7AX-PTD-UTM *
23	3.6736E-07	RA-1D * EFWOP7AX-PTD-LF * RA-VCCD * SWS3806A-VCC-LF * LPI1408B-VCC-LF *
24	3.6736E-07	RA-1D * EFWOP7AX-PTD-LF * RA-VCCD * SWS3810B-VCC-LF * SWS3806A-VCC-LF *
25	3.6736E-07	RA-1D * EFWOP7AX-PTD-LF * RA-VCCD * SWS3806A-VCC-LF * SWS3807B-VCC-LF *
26	3.6540E-07	RA-1D * EFWOP7AX-PTD-LF * FNB-MANACT * EFWOP7BA-PMO-LF *
27	2.8011E-07	RA-1D * EFWOP7AX-PTD-LF * RA-VCCD * SWS3807B-VCC-LF * DG1-GEN-UTM *
28	2.8011E-07	RA-1D * EFWOP7AX-PTD-LF * RA-VCCD * LPI1408B-VCC-LF * DG1-GEN-UTM *
29	2.8011E-07	RA-1D * EFWOP7AX-PTD-LF * RA-VCCD * SWS3810B-VCC-LF * DG1-GEN-UTM *
30	2.8011E-07	RA-1D * EFWOP7AX-PTD-LF * RA-VCCD * SWS3806A-VCC-LF * DG2-GEN-UTM *
31	2.5620E-07	RA-1D * EFWOP7AX-PTD-LF * DG1-GEN-UTM * FNB-MANACT *
32	1.7539E-07	RA-1D * EFWOP7AX-PTD-LF * HPIP36CB-PMO-LF * BATTD07-ZBT-LF * RA-9D *
184	1.3640E-09	RA-1D * EFWOP7AX-PTD-LF * LF-SWS-VCH4B * RA-7G * HPISV-CM *
185	9.8560E-10	RA-1D * EFWOP7AX-PTD-LF * RA-7G * LF-ECS-B6 * BATTD06-ZBT-UTM *
186	9.8192E-10	RA-1D * LF-SWS-VCH4B * RA-7G * LF-SWS-VCH4A * EFWOP7AX-PTD-UTM *
187	7.5240E-10	RA-1D * LF-SWS-VCH4B * RA-7G * BATTD06-ZBT-LF * LF-REF-RM109 *
188	7.5240E-10	RA-1D * RA-7G * LF-SWS-VCH4A * BATTD07-ZBT-LF * LF-REF-RM95

THE MAXIMUM TERM VALUE FOR COMPUTATION 1 IS 1.760000000000E-04
 (THE SUM OF THE TERM VALUES IS 2.535527096959E-04)

TERM
NUMBER

PROP.
OF TERM

Table B.22 ANO-1 Boolean Accident Sequences After
Recovery Has Been Applied (continued)

ANOR30 =

1	8.2215E-07	EFWOP7AX-PTD-LF * RA-2D * FNB-MANACT * EFWOP7BA-PMD-LF *
2	3.1500E-07	EFWOP7AX-PTD-LF * RA-2D * SWSP-CM *
3	2.7000E-07	RA-2D * FNB-MANACT * EFWSP-CM *
4	2.7000E-07	RA-2D * FNB-MANACT * EFWC019X-XUC-LF *
5	1.5840E-07	RA-2D * LOSS-OSP * BATT-CM * RA-1D * RA-8D *
6	1.1372E-07	LF-SWS-VCH4A * LF-SWS-VCH4B * EFWOP7AX-PTD-LF * RA-7G * RA-2D *
7	5.2069E-08	LF-SWS-VCH4B * EFWOP7AX-PTD-LF * RA-7G * RA-2D * HPIP36CB-PMD-LF *
8	4.7880E-08	LF-SWS-VCH4B * EFWOP7AX-PTD-LF * RA-7G * RA-2D * LPI1408B-VCC-LF *
9	4.7880E-08	LF-SWS-VCH4B * EFWOP7AX-PTD-LF * RA-7G * RA-2D * SWS3810B-VCC-LF *
10	4.4944E-08	EFWOP7AX-PTD-LF * RA-2D * EFWOP7BA-PMD-LF * HPISV-CM * RA-VCCD *
11	3.8304E-08	LF-SWS-VCH4A * EFWOP7AX-PTD-LF * RA-7G * RA-2D * LF-ECS-B6 *
12	3.4114E-08	LF-SWS-VCH4B * EFWOP7AX-PTD-LF * RA-7G * RA-2D * SWSOP4CB-PMD-LF *
13	3.4020E-08	EFWOP7AX-PTD-LF * RA-2D * SWS3824-VOC-LF * RA-6D *
14	3.0703E-08	EFWOP7AX-PTD-LF * RA-2D * RA-6D * SWSOP4CB-PAC-LF * SWSOP4BA-PAC-LF *
15	2.8728E-08	LF-SWS-VCH4B * EFWOP7AX-PTD-LF * RA-7G * RA-2D * LF-ECS-A6 *
16	1.7955E-08	LF-SWS-VCH4B * EFWOP7AX-PTD-LF * RA-7G * RA-2D * FNB-MANACT *
17	1.7539E-08	EFWOP7AX-PTD-LF * RA-7G * RA-2D * HPIP36CB-PMD-LF * LF-ECS-B6 *
18	1.6200E-08	RA-2D * FNB-MANACT * RA-6D * TM-LOGIC-CM *
19	1.6128E-08	EFWOP7AX-PTD-LF * RA-7G * RA-2D * LPI1408B-VCC-LF * LF-ECS-B6 *
20	1.6128E-08	EFWOP7AX-PTD-LF * RA-7G * RA-2D * SWS3810B-VCC-LF * LF-ECS-B6 *

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Table B.22 ANO-1 Boolean Accident Sequences After
Recovery Has Been Applied (continued)

21	1.1491E-08	EFWOP7AX-PTD-LF * RA-7G * RA-2D * LF-ECS-B6 * SWSOP4CB-PMD-LF *
22	9.5768E-09	EFWOP7AX-PTD-LF * RA-7G * RA-2D * LF-ECS-B6 * LF-ECS-A6 *
23	6.0480E-09	EFWOP7AX-PTD-LF * RA-7G * RA-2D * LF-ECS-B6 * FNB-MANACT *
24	2.3940E-09	LF-SWS-VCH4B * EFWOP7AX-PTD-LF * RA-7G * RA-2D * SWSV-08-10-CM *
25	2.3940E-09	LF-SWS-VCH4B * EFWOP7AX-PTD-LF * RA-7G * RA-2D * HPISV-CM *
26	2.2093E-09	LF-SWS-VCH4A * LF-SWS-VCH4B * RA-7G * RA-2D * EFWOP7AX-PTD-UTM *
27	1.6758E-09	LF-SWS-VCH4B * EFWOP7AX-PTD-LF * RA-7G * RA-2D * HPIP36CB-PMD-UTM *
28	1.4296E-09	LF-SWS-VCH4A * LF-SWS-VCH4B * RA-7G * RA-2D * BATT-CM *
29	1.0116E-09	LF-SWS-VCH4B * RA-7G * RA-2D * HPIP36CB-PMD-LF * EFWOP7AX-PTD-UTM *
30	9.3324E-10	LF-SWS-VCH4B * RA-7G * RA-2D * LPI1408B-VCC-LF * EFWOP7AX-PTD-UTM *
31	9.3324E-10	LF-SWS-VCH4B * RA-7G * RA-2D * SWS3810B-VCC-LF * EFWOP7AX-PTD-UTM

THE MAXIMUM TERM VALUE FOR COMPUTATION 1 IS 8.221500000000E-07
(THE SUM OF THE TERM VALUES IS 2.432049246000E-06)

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TERM
NUMBER

PROR.
OF TERM

Table B.22 ANO-1 Boolean Accident Sequences After
Recovery Has Been Applied (continued)

ANOR30A =

1	9.1350E-07	EFWOP7AX-PTD-LF * FNB-MANACT * EFWOP7BA-PMD-LF +
2	3.5000E-07	EFWOP7AX-PTD-LF * SWSP-CM +
3	3.0000E-07	FNB-MANACT * EFWSP-CM +
4	3.0000E-07	FNB-MANACT * EFW0019X-XOC-LF +
5	1.7600E-07	LOSS-OSP * BATT-CM * RA-1D * RA-8D +
6	1.2635E-07	LF-SWS-VCH4A * LF-SWS-VCH4B * EFWOP7AX-PTD-LF * RA-7G +
7	5.7855E-08	LF-SWS-VCH4B * EFWOP7AX-PTD-LF * RA-7G * HPIP36CB-PMD-LF +
8	5.3200E-08	LF-SWS-VCH4B * EFWOP7AX-PTD-LF * RA-7G * LPI1408B-VCC-LF +
9	5.3200E-08	LF-SWS-VCH4B * EFWOP7AX-PTD-LF * RA-7G * SWS3810B-VCC-LF +
10	4.9938E-08	EFWOP7AX-PTD-LF * EFWOP7BA-PMD-LF * HPISV-CM * RA-VCCD +
11	4.2560E-08	LF-SWS-VCH4A * EFWOP7AX-PTD-LF * RA-7G * LF-ECS-B6 +
12	3.7905E-08	LF-SWS-VCH4B * EFWOP7AX-PTD-LF * RA-7G * SWSOP4CB-PMD-LF +
13	3.7800E-08	EFWOP7AX-PTD-LF * SWS3824-VOC-LF * RA-6D +
14	3.4114E-08	EFWOP7AX-PTD-LF * RA-6D * SWSOP4CB-PAC-LF * SWSOP4BA-PAC-LF +
15	3.1920E-08	LF-SWS-VCH4B * EFWOP7AX-PTD-LF * RA-7G * LF-ECS-A6 +
16	1.9950E-08	LF-SWS-VCH4B * EFWOP7AX-PTD-LF * RA-7G * FNB-MANACT +
17	1.9488E-08	EFWOP7AX-PTD-LF * RA-7G * HPIP36CB-PMD-LF * LF-ECS-B6 +
18	1.8000E-08	FNB-MANACT * RA-6D * TM-LOGIC-CM +
19	1.7920E-08	EFWOP7AX-PTD-LF * RA-7G * LPI1408B-VCC-LF * LF-ECS-B6 +
20	1.7920E-08	EFWOP7AX-PTD-LF * RA-7G * SWS3810B-VCC-LF * LF-ECS-B6 +

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Table B.22 ANO-1 Boolean Accident Sequences After
Recovery Has Been Applied (continued)

21	1.2768E-08	EFWOP7AX-PTD-LF * RA-7G * LF-ECS-B6 * SWSOP4CB-PMD-LF *
22	1.0752E-08	EFWOP7AX-PTD-LF * RA-7G * LF-ECS-B6 * LF-ECS-A6 *
23	6.7200E-09	EFWOP7AX-PTD-LF * RA-7G * LF-ECS-B6 * FNB-MANACT *
24	2.6600E-09	LF-SWS-VCH4B * EFWOP7AX-PTD-LF * RA-7G * SWSV-08-10-CH *
25	2.6600E-09	LF-SWS-VCH4B * EFWOP7AX-PTD-LF * RA-7G * HPISV-CH *
26	2.4548E-09	LF-SWS-VCH4A * LF-SWS-VCH4B * RA-7G * EFWOP7AX-PTD-UTM *
27	1.8620E-09	LF-SWS-VCH4B * EFWOP7AX-PTD-LF * RA-7G * HPIP36CE-PMD-UTM *
28	1.5884E-09	LF-SWS-VCH4A * LF-SWS-VCH4B * RA-7G * BATT-CH *
29	1.1240E-09	LF-SWS-VCH4B * RA-7G * HPIP36CB-PMD-LF * EFWOP7AX-PTD-UTM *
30	1.0336E-09	LF-SWS-VCH4B * RA-7G * LPI1408B-VCC-LF * EFWOP7AX-PTD-UTM *
31	1.0336E-09	LF-SWS-VCH4B * RA-7G * SWS3810B-VCC-LF * EFWOP7AX-PTD-UTM

THE MAXIMUM TERM VALUE FOR COMPUTATION 1 IS 9.135000000000E-07
(THE SUM OF THE TERM VALUES IS 2.702276940000E-06)

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TERM
NUMBER

PROB.
OF TERM

Table B.22 ANO-1 Boolean Accident Sequences After
Recovery Has Been Applied (continued)

ANOR38 =

1	2.7405E-04	EFWOP7AX-PTD-LF * RA-2D * HPIP36CB-PMD-LF +
2	1.0332E-04	EFWOP7AX-PTD-LF * RA-2D * SWS3810B-VCC-LF * RA-VCCD +
3	1.0332E-04	EFWOP7AX-PTD-LF * RA-2D * RA-VCCD * LPI1408B-VCC-LF +
4	9.4500E-05	EFWOP7AX-PTD-LF * RA-2D * FNB-MANACT +
5	8.9200E-06	EFWOP7AX-PTD-LF * RA-2D * HPIP36CB-PMD-UTM +
6	5.9850E-06	LF-SWS-VCH4A * EFWOP7AX-PTD-LF * RA-2D * RA-7G +
7	5.3865E-06	EFWOP7AX-PTD-LF * RA-2D * SWSOP4CB-PMD-LF * RA-6D +
8	5.3244E-06	RA-2D * HPIP36CB-PMD-LF * EFWOP7AX-PTD-UTM +
9	5.1661E-06	EFWOP7AX-PTD-LF * RA-2D * RA-VCCD * SWSV-08-10-CM +
10	5.1660E-06	EFWOP7AX-PTD-LF * RA-2D * RA-VCCD * HPISV-CM +
11	3.4650E-06	EFWOP7AX-PTD-LF * RA-2D * HPIU18CX-XOC-UTM +
12	3.4650E-06	EFWOP7AX-PTD-LF * RA-2D * SWS002CX-XOC-UTM +
13	3.4650E-06	EFWOP7AX-PTD-LF * RA-2D * SWS018CX-XOC-UTM +
14	3.4650E-06	EFWOP7AX-PTD-LF * RA-2D * HPI08W2X-XOC-UTM +
15	3.4650E-06	EFWOP7AX-PTD-LF * RA-2D * SWS3641B-VOC-UTM +
16	3.4650E-06	EFWOP7AX-PTD-LF * RA-2D * HPIU20CX-XOC-UTM +
17	3.1500E-06	EFWOP7AX-PTD-LF * RA-2D * LPIBW1X-XOC-LF +
18	3.1500E-06	EFWOP7AX-PTD-LF * RA-2D * HPIU18CX-XOC-LF +
19	3.1500E-06	EFWOP7AX-PTD-LF * RA-2D * HPIU20CX-XOC-LF +
20	3.1500E-06	EFWOP7AX-PTD-LF * RA-2D * HPIU19CX-CCC-LF +

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Table B.22 ANO-1 Boolean Accident Sequences After
Recovery Has Been Applied (continued)

21	3.1500E-06	EFWOP7AX-PTD-LF * RA-2D * SWS018CX-XOC-LF *
22	3.1500E-06	EFWOP7AX-PTD-LF * RA-2D * HPI08W2X-XOC-LF *
23	2.6460E-06	EFWOP7AX-PTD-LF * RA-2D * B6-480VAC-LF *
24	2.0160E-06	EFWOP7AX-PTD-LF * RA-2D * HPI1228B-VCC-LF * HPI1227B-VCC-LF *
25	2.0074E-06	RA-2D * RA-VCCD * LPI1408B-VCC-LF * EFWOP7AX-PTD-UTM *
26	2.0074E-06	RA-2D * SWS38103-VCC-LF * RA-VCCD * EFWOP7AX-PTD-UTM *
27	1.8907E-06	EFWOP7AX-PTD-LF * RA-2D * A4-4160VAC-LF *
28	1.8350E-06	RA-2D * FNH-MANACT * EFWOP7AX-PTD-UTM *
29	1.5120E-06	EFWOP7AX-PTD-LF * RA-2D * RA-7G * LF-ECS-A6 *
30	1.2915E-06	EFWOP7AX-PTD-LF * RA-2D * RA-VCCD * HPSV-CM *
31	1.1340E-06	EFWOP7AX-PTD-LF * RA-2D * SWS3641B-VOC-LF *
32	9.4500E-07	EFWOP7AX-PTD-LF * RA-2D * 002-125VDC-LF *
159	2.4624E-09	LF-SWS-VCH4A * RA-2D * RA-7G * EFW00Y2B-VCC-LF * BATTOG7-ZBT-LF *
170	2.2572E-09	LF-SWS-VCH4A * RA-2D * RA-7G * LF-REF-RM95 * LF-DC-Y11 *
171	2.0520E-09	LF-SWS-VCH4A * RA-2D * RA-7G * ACT-EFAA-LF * LF-DC-Y22 *
172	2.0520E-09	LF-SWS-VCH4A * RA-2D * RA-7G * ACT-EFBB-LF * LF-DC-Y11 *
173	1.7280E-09	RA-2D * RA-7G * LF-ECS-A6 * EFW00Y1A-VCC-LF * ACT-EFBB-LF

THE MAXIMUM TERM VALUE FOR COMPUTATION 1 IS 2.740500000000E-04
(THE SUM OF THE TERM VALUES IS 6.907541395498E-04)

TERM
NUMBER

PROB.
OF TERM

Table B.22 ANO-1 Boolean Accident Sequences After
Recovery Has Been Applied (continued)

ANOR46 =

1	2.7405E-04	EFWOP7AX-PTD-LF * RA-2D * HPIP36CB-PMD-LF *
2	1.0332E-04	EFWOP7AX-PTD-LF * RA-2D * SWS3810B-VCC-LF * RA-VCCD *
3	1.0332E-04	EFWOP7AX-PTD-LF * RA-2D * RA-VCCD * LPI1408B-VCC-LF *
4	9.4511E-05	EFWOP7AX-PTD-LF * RA-2D * FNB-MANACT *
5	2.7000E-05	RA-2D * D02-125VDC-LF *
6	2.7000E-05	RA-2D * D21-125VDC-LF *
7	2.5692E-05	RA-2D * HPIP36CB-PMD-LF * RA-VCCD * EFW00Y2B-VCC-LF *
8	2.3616E-05	RA-2D * SWS3810B-VCC-LF * RA-VCCD * EFW00Y2B-VCC-LF *
9	2.3616E-05	RA-2D * RA-VCCD * LPI1408B-VCC-LF * EFW00Y2B-VCC-LF *
10	8.8560E-06	RA-2D * RA-VCCD * EFW00Y2B-VCC-LF * FNB-MANACT *
11	8.8200E-06	EFWOP7AX-PTD-LF * RA-2D * HPIP36CB-PMD-UTM *
12	5.9850E-06	LF-SWS-VCH4A * EFWOP7AX-PTD-LF * RA-2D * RA-7G *
13	5.3865E-06	EFWOP7AX-PTD-LF * RA-2D * SWSOP4CB-PMD-LF * RA-6D *
14	5.3244E-06	RA-2D * HPIP36CB-PMD-LF * EFWOP7AX-PTD-UTM *
15	5.1660E-06	EFWOP7AX-PTD-LF * RA-2D * RA-VCCD * HPISV-CM *
16	5.1660E-06	EFWOP7AX-PTD-LF * RA-2D * RA-VCCD * SWSV-08-10-CM *
17	3.4650E-06	EFWOP7AX-PTD-LF * RA-2D * HPIU18CX-XOC-UTM *
18	3.4650E-06	EFWOP7AX-PTD-LF * RA-2D * HPIU20CX-XOC-UTM *
19	3.4650E-06	EFWOP7AX-PTD-LF * RA-2D * SWS002CX-XOC-UTM *
20	3.4650E-06	EFWOP7AX-PTD-LF * RA-2D * SWS018CX-XOC-UTM *

Table B.22 ANO-1 Boolean Accident Sequences After
Recovery Has Been Applied (continued)

21	3.4650E-06	EFWOP7AX-PTD-LF * RA-2D * HPI0BW2X-XOC-UTH +
22	3.4650E-06	EFWOP7AX-PTD-LF * RA-2D * SWS3641B-VOC-UTH +
23	3.1500E-06	EFWOP7AX-PTD-LF * RA-2D * HPIU18CX-XOC-LF +
24	3.1500E-06	EFWOP7AX-PTD-LF * RA-2D * HPIU20CX-XOC-LF +
25	3.1500E-06	EFWOP7AX-PTD-LF * RA-2D * LPIHW1X-XOC-LF +
26	3.1500E-06	EFWOP7AX-PTD-LF * RA-2D * HPI0BW2X-XOC-LF +
27	3.1500E-06	EFWOP7AX-PTD-LF * RA-2D * HPIU19CX-CCC-LF +
28	3.1500E-06	EFWOP7AX-PTD-LF * RA-2D * SWS018CX-XOC-LF +
29	2.6460E-06	EFWOP7AX-PTD-LF * RA-2D * B6-480VAC-LF +
30	2.0074E-06	RA-2D * RA-VCCD * LPI1408B-VCC-LF * EFWOP7AX-PTD-UTH +
31	2.0074E-06	RA-2D * SWS3810B-VCC-LF * RA-VCCD * EFWOP7AX-PTD-UTH +
32	1.8900E-06	EFWOP7AX-PTD-LF * RA-2D * A4-4160VAC-LF +
231	4.3200E-09	RA-2D * RA-7G * LF-ECS-A6 * LF-DC-Y22DC +
232	4.3200E-09	RA-2D * RA-7G * LF-ECS-A6 * EFW0019X-XOC-LF +
233	4.3200E-09	RA-2D * RA-7G * LF-ECS-A6 * EFWSV-CM +
234	4.3200E-09	RA-2D * RA-7G * LF-ECS-A6 * EFW010BX-CCC-LF +
235	4.3200E-09	RA-2D * RA-7G * LF-ECS-A6 * EFWSV-CM

THE MAXIMUM TERM VALUE FOR COMPUTATION 1 IS 2.740500000000E-04
(THE SUM OF THE TERM VALUES IS 8.476759435496E-04)

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Table B.23 ANO-1 Dominant Cut Sets by
Accident Sequence with Recovery

Anal. No.	Boolean Event No.	Core Melt Failure Sequence	Cut Set Description	Cut Set Core Melt Probability	Vulner- ability No.
8	9	$S_3MH_1H_2'$	SWSV-40-41-CM	3.04E-6	2
			LPRSV-CM	3.04E-6	2
			SWSV-02-03-CM	3.04E-6	2
			LPIOP34A-PMD-LF*		
			LPIOP34B-PMD-LF	1.51E-6	6
			LPSP-CM	1.40E-6	3
			LPIOP34A-PMD-LF or LPIOP34B-PMD-LF*		
			Various Singles (51 cut sets)	5.11E-6	6
			LPI1407A-VCC-LF or LPI1408B-VCC-LF*		
			Various Singles (6 cut sets)	1.11E-6	8
			HPRS-MANACT*		
			LPRS-MANACT	1.80E-7	7
			All other cut sets	<u>3.96E-6</u>	
10	11	S_3MED_2		2.34E-5	
			HPISV-CM	2.96E-6	2
			LPIBWIX-XOC-LF	1.90E-6	8
			LPI1407A-VCC-LF or LPI1408B-VCC-LF*		
			Various Singles (3 cut sets)	5.73E-7	8
			SWSP-CM	1.90E-7	3
			All other cut sets	<u>1.01E-6</u>	
				6.60E-6	

Table B.23 ANO-1 Dominant Cut Sets by Accident Sequence with Recovery (continued)

Anal. No.	Boolean Event No.	Core Melt Failure Sequence	Cut Set Description	Cut Set Core Melt Probability	Vulnerability No.
11	12	S ₃ MXE	FNB-MANACT	1.98E-6	7
			HPISV-CM	1.03E-7	2
			LPIBWIX-XOC-LF	6.60E-8	8
			HPSV-CM	2.57E-8	2
			HPSP-CM	1.98E-8	3
			LPI1407A-VCC-LF* Various Singles (5 cut sets)	5.29E-8	8
			SWSP-CM	2.57E-9	3
			All other cut sets	<u>7.64E-8</u>	
				2.33E-6	
15	22	T ₁ MLE	BATT-CM	1.48E-5	5
			DG-CM*EFWOP7AX-PTD-LF	1.76E-6	4
			DG1-GEN-LF*DG2-GEN-LF* EFWOP7AX-PTD-LF	1.70E-6	4
			DG1-GEN-LF or DG2-GEN-LF* EFWOP7AX-PTD-LF* Various Singles (10 cutsets)	1.59E-6	4
			EFWOP7AX-PTD-LF* SWSV-06-07-CM	1.93E-7	1
			BATTD07-ZBT-LF* BATTD26-ZBT-LF	8.71E-8	5
			All other cut sets	<u>1.16E-6</u>	
				2.13E-5	

Table B.23 ANO-1 Dominant Cut Sets by Accident Sequence with Recovery (continued)

Anal. No.	Boolean Event No.	Core Melt Failure Sequence	Cut Set Description	Cut Set Core Melt Probability	Vulnerability No.
17	30	T ₂ MLE	EFWOP7AX-PTD-LF* FNB-MANACT* EFWOP7BA-PMD-LF	8.06E-7	7
			EFWOP7AX-PTD-LF* SWSP-CM	3.09E-7	1
			FNB-MANACT* Various Singles (3 cut sets)	5.45E-7	7
			BATT-CM*LOSS-OSP	1.55E-7	5
			EFWOP7AX-PTD-LF* Various Doubles (19 cut sets)	5.29E-7	1
			EFWOP7AX-PTD-LF* SWS3824-VOC-LF	3.33E-8	1
			All other cut sets	<u>6.30E-9</u> 2.38E-6	
19	30a	T ₃ MLE	EFWOP7AX-PTD-LF* FNB-MANACT* EFWOP7BA-PMD-LF	3.84E-7	7
			EFWOP7AX-PTD-LF* SWSP-CM	1.47E-7	1
			FNB-MANACT* Various Singles (3 cut sets)	2.42E-7	7
			BATT-CM*LOSS-OSP	7.39E-8	5
			EFWOP7AX-PTD-LF* Various Doubles (19 cut sets)	2.52E-7	1
			EFWOP7AX-PTD-LF* SWS3824-VOC-LF	1.59E-8	1
			All other cut sets	<u>3.00E-9</u> 1.14E-6	

Table B.23 ANO-1 Dominant Cut Sets by Accident
Sequence with Recovery (continued)

Anal. No.	Boolean Event No.	Core Melt Failure Sequence	Cut Set Description	Cut Set Core Melt Probability	Vulner- ability No.
26	21	T_1MQLD_1	BATT-CM	5.24E-7	5
			DG-CM*EFWOP7AX-PTD-LF	6.25E-8	4
			DG1-GEN-LF*DG2-GEN-LF* EFWOP7AX-PTD-LF	6.01E-8	4
			DG1-GEN-LF or DG2-GEN-LF* EFWOP7AX-PTD-LF* Various Singles (14 cut sets)	7.47E-8	4
			EFWOP7AX-PTD-LF* SWS-06-07-CM	9.33E-9	1
			BATTD07-ZBT-LF* BATTD06-ZBT-LF	3.86E-9	5
			All other cut sets	<u>5.04E-8</u> 7.85E-7	
27	1	$T_2MQH_1H'_2$	SWSV-40-41-CM	2.13E-7	2
			LPRSV-CM	2.13E-7	2
			SWSV-02-03-CM	2.13E-7	2
			LPIOP34A-PMD-LF* LPIOP34B-PMD-LF	1.06E-7	6
			LPSP-CM	9.80E-8	3
			LPIOP34A-PMD-LF or LPIOP34B-PMD-LF* Various Singles (53 cut sets)	3.60E-7	6
			LPI1407A-VCC-LF or LPI1408B-VCC-LF* Various Singles (6 cut sets)	7.76E-8	3
			HPRS-MANACT*LPRS-MANACT or HPRS-MANACT*LPSV-CM	1.43E-8	7
			All other cut sets	<u>4.12E-7</u> 1.71E-6	

Table B.23 ANO-1 Dominant Cut Sets by Accident
Sequence with Recovery (continued)

Anal. No.	Boolean Event No.	Core Melt Failure Sequence	Cut Set Description	Cut Set Core Melt Probability	Vulner- ability No.
29	3a	$T_2MQD_1D_2$	HPISV-CM	3.14E-7	2
			LPIBWIX-XOC-LF	1.40E-7	8
			LPI1407A-VCC-LF or LPI1408B-VCC-LF* Various Singles (4 cut sets)	5.78E-8	8
			SWSP-CM	1.40E-8	3
			All other cut sets	<u>1.41E-7</u>	
				6.66E-7	
33	1	$T_3MQH_1H_2'$	SWSV-40-41-CM	9.12E-8	2
			LPRSV-CM	9.12E-8	2
			SWSV-02-03-CM	9.12E-8	2
			LPIOP34A-PMD-LF*		
			LPIOP34B-PMD-LF	4.54E-8	6
			LPSP-CM	4.20E-8	3
			LPIOP34A-PMD-LF or LPIOP34B-PMD-LF* Various Singles (53 cut sets)	1.54E-7	6
			LPI1407A-VCC-LF or LPI1408B-VCC-LF* Various Singles (6 cut sets)	3.33E-8	8
			HPRS-MANACT*LPRS-MANACT or HPRS-MANACT*LPSV-CM	6.12E-9	7
			All other cut sets	<u>1.76E-7</u>	
				7.31E-7	

Table B.23 ANO-1 Dominant Cut Sets by Accident
Sequence with Recovery (continued)

Anal. No.	Boolean Event No.	Core Melt Failure Sequence	Cut Set Description	Cut Set Core Melt Probability	Vulner- ability No.
39	1	T ₃ QH ₁ H ₂ '	SWSV-40-51-CM	1.51E-6	2
			LPRSV-CM	1.51E-6	2
			SWSV-02-03-CM	1.51E-6	2
			LPIOP34A-PMD-LF*		
			LPIOP34B-PMD-LF	7.49E-7	6
			LPSP-CM	6.93E-7	3
			LPIOP34A-PMD-LF or LPIOP34B-PMD-LF*		
			Various Singles (53 cut sets)	2.55E-6	6
			LPI1407A-VCC-LF or LPI1408B-VCC-LF*		
			Various Singles (6 cut sets)	5.49E-7	8
41	3a	T ₃ QD ₁ D ₂	HPRs-MANACT*LPRs-MANACT or HPRs-MANACT*LPSV-CM	1.01E-7	7
			All other cut sets	<u>2.91E-6</u>	
				1.21E-5	
			HPISV-CM	2.15E-6	2
			LPIBWIX-XOC-LF	9.60E-7	8
			LPI1407A-VCC-LF or LPI1408B-VCC-LF*		
			Various Singles (4 cut sets)	3.96E-7	8
			SWSP-CM	9.60E-8	3
			All other cut sets	<u>9.67E-7</u>	
				4.57E-6	

Table B.23 ANO-1 Dominant Cut Sets by Accident
Sequence with Recovery (continued)

Anal. No.	Boolean Event No.	Core Melt Failure Sequence	Cut Set Description	Cut Set Core Melt Probability	Vulner- ability No.
51	38	T ₄ MLE	EFWOP7AX-PTD-LF* Various Singles and Doubles (30 cut sets)	3.28E-6	1
			EFWOP7AX-PTD-LF/UTM* FNB-MANACT	5.68E-7	7
			EFWOP7AX-PTD-UTM* Various Singles (2 cut sets)	5.51E-8	1
			EFWOP7AX-PTD-LF* SWSV-08-10-CM	3.05E-8	1
			EFWOP7AX-PTD-LF* HPISV-CM	3.05E-8	1
			EPWOP7AX-PTD-LF* HPSV-CM	7.62E-9	1
			All other cut sets	<u>1.08E-7</u> 4.08E-6	
59	46	T ₅ MLE	EFWOP7AX-PTD-LF* Various Singles (21 cut sets)	1.59E-6	1
			EFWOP7AX-PTD-LF/UTM* FNB-MANACT	2.79E-7	7
			EFWOP7AX-PTD-UTM* Various Singles (3 cut sets)	2.71E-8	1
			EFW00Y2B-VCC-LF* Various Singles (4 cut sets)	2.37E-7	1
			EFWOP7AX-PTD-LF* HPISV-CM	1.50E-8	1
			EFWOP7AX-PTD-LF* SWSV-08-10-CM	1.50E-8	1
			All other cut sets	<u>3.00E-7</u> 2.46E-6	

accident sequence contributed by all the other cut sets in that accident sequence that are not explicitly considered further.

The next step is to sort all the most significant cut sets identified for the top 14 accident sequences and sum the total core melt probability for each cut set over all the accident sequences to which it contributes. This is given in Table B.24. These cut sets are the vulnerabilities described in Section 2 of the main report and subsequently communicated to the architectural engineer and discussed at the second plant visit along with those vulnerabilities identified for the special emergencies.

The internal analysis started with 60 accident sequences with a total core melt probability of $4.72\text{E-}4$ before recovery was applied. The dominant cut sets accounted for $4.66\text{E-}4$ of the total core melt probability. In order to figure the residual core melt probability that is not being addressed consider the following:

- 1) The initial core melt probability before recovery was applied for all 60 accident sequences was $4.7204\text{E-}4$.
- 2) There were 37 accident sequences not significant enough to apply recovery factors other than RA-1 or RA-2 (see Table B.4).
- 3) The core melt probability of the 60 accident sequences including 23 with full recovery, 16 with partial recovery, and 21 with no recovery applied is $8.7578\text{E-}5$.
- 4) The 46 accident sequences not carried further in the analysis constitute the first residual of $4.39\text{E-}6$.
- 5) When the top 14 accident sequences were analyzed, eight significant vulnerabilities were identified with a core melt probability of $7.1913\text{E-}5$.
- 6) Those cut sets from the top 14 accident sequences which were not associated with one of the eight vulnerabilities amounted to the second residual of $1.1277\text{E-}5$.
- 7) Thus the total residual core melt probability including sequences and cut sets not significant or identifiable as specific vulnerabilities amounts to $1.567\text{E-}5$.
- 8) This residual core melt probability would really be lower if recovery had been applied to all accident sequences and all cut sets in the dominant accident sequences, and if other cut sets of lower probability had been associated with the eight vulnerabilities. The improvement could be a factor of approximately 2 to 4.

Table B.24 ANO-1 Vulnerabilities

Vulnerability	Core Melt Probability Contribution
1 EFWOP7AX-PTD-LF * Various Singles and Doubles	6.77E-6
2 VALVE-CM	2.01E-5
3 PUMP-CM	2.56E-6
4 DG-CM * EFWOP7AX-PTD-LF + DG1-GEN-LF * DG2-GEN-LF * EFWOP7AX-PTD-LF + DG1-GEN-LF or DG2-GEN-LF * EFWOP7AX-PTD-LF * Various Singles	5.25E-6
5 BATT-CM + BATT-CM * LOSS-OSP + BATTD07-ZBT-LF * BATTD06-ZBT-LF	1.56E-5
6 LPIOP34A-PMD-LF * LPIOP34B-PMD-LF + LPIOP34A-PMD-LF or LPIOP34B-PMD-LF * Various Singles	1.06E-5
7 FNB-MANACT + FNB-MANACT * EFWOP7AX-PTD-LF * EFWD07BA-PMD-LF + FNB-MANACT * Various Singles + HPRS-MANACT * LPRS-MANACT + HPRS-MANACT * Various Singles	5.11E-6
8 LPIBWIX-XOC-LF + LPI1407A-VCC-LF * Various Singles + LPI1408B-VCC-LF * Various Singles	5.92E-6
Residual accident sequences and cut sets after recovery was applied	<u>1.57E-5</u>
Total core melt probability for internal analysis after recovery was applied	8.76E-5

5.5 Containment Systems Integration

In order to determine public risk measures, the core melt accident sequences must be expanded to include containment systems failure and then analyzed for containment failure modes, assigned release categories, and propagated in a consequence analysis. Therefore, the next step was to apply the containment event tree to each of the 14 dominate accident sequences.

The containment systems event tree is shown in Figure B.5 in a previous subsection of this Appendix. Each of the fourteen dominant core melt accident sequences had to be rerun with each containment system sequence. The outcomes of the containment systems event tree cover all the combinations of success or failure of containment overpressure protection (COP) and post accident radioactivity removal (PARR). It should be noted that even the containment system success sequence $\bar{Z}\bar{C}\bar{F}$ is important since the containment will fail by some means not identified by the containment systems event tree given a core melt.

The matrix of the SETS runs is given in Table B.25. Each distinctly different Boolean accident sequence (there are 10 before recovery) corresponding to the 14 dominant accident sequences was run with all the Boolean containment system sequences. This is necessary due to the dependencies between systems. For example, consider the analysis sequence number 15 which corresponds to ANO sequence number 22. The multiplier expression is $T_1MP\bar{Q}$ and the Boolean events are LE. Thus LE must be run with each containment sequence, i.e.,

LE $\bar{Z}\bar{C}\bar{F}$
LE $\bar{Z}\bar{C}F$
LE ZC
LE $Z\bar{C}\bar{F}$
LE $Z\bar{C}F$, and
LE ZC .

The corresponding probabilities would be $p(T_1MP\bar{Q}) * p(LE\bar{Z}\bar{C}\bar{F})$ etc. It would be incorrect to assume that

$$p(LE\bar{Z}\bar{C}\bar{F}) = p(LE)p(\bar{Z}\bar{C}\bar{F}).$$

After the SETS runs were completed the resulting probabilities for the total accident sequence were screened to determine those sequences to be carried on to the recovery analysis and placed into release categories. The results before recovery are given in Table B.26 and after recovery in Table B.27. The next step in the analysis leading to public risk measures can be found in Section 7.2 of the main report.

Table B.25 ANO-1 Containment Sequences Analyzed

LOCA

ANO No. (Boolean)	$\bar{Z}_L \bar{C} \bar{F}$	$\bar{Z}_L \bar{C} F$	$\bar{Z}_L C$	$Z_L \bar{C} \bar{F}$	$Z_L \bar{C} F$	$Z_L C$
1	*	*	*	*	*	*
3	*	*	*	*	*	*
9	*	*	*	*	*	*
11	*	*	*	*	*	*
12	*	*	*	*	*	*
21	*	*	*	*	*	*

where $Z_L = Y(C + F + H_1)$

TRANSIENT

ANO No. (Boolean)	$\bar{Z}_T \bar{C} = \bar{C}$	$\bar{Z}_T C = \bar{Y}C$	$Z_T = YC$
22	*	*	*
30	*	*	*
38	*	*	*
46	*	*	*

where $Z_T = YC$

Table B.26 Total Accident Sequence Probabilities Including Containment
Systems Failures Before Recovery Considered - Base Case

Core Melt Accident Sequence	Analysis Number	ANO Boolean Sequence Number	Containment System Sequences					
			$\bar{Z}_L \bar{C} \bar{F}$	$\bar{Z}_L \bar{C} F$	$\bar{Z}_L C$	$Z_L \bar{C} \bar{F}$	$Z_L \bar{C} F$	$Z_L C$
			* $\bar{Z}_T \bar{C}$		$\bar{Z}_T C$			Z_T
$S_3^{MH_1 H'_2}$	8	9	6.73E-5	4.76E-5	2.53E-6	7.17E-7	7.32E-7	7.75E-9
$S_3^{MED_2}$	10	11	5.60E-7	ϕ	2.25E-5	ϕ	ϕ	3.16E-5
S_3^{MXE}	11	12	2.68E-6	2.06E-8	8.05E-7	1.22E-8	1.13E-10	1.10E-6
T_1^{MLE} *	15	22	5.15E-6	--	2.03E-6	--	--	6.55E-5
T_2^{MLE} *	17	30	1.99E-5	--	7.45E-6	--	--	2.93E-5
T_3^{MLE} *	19	30a	8.52E-6	--	3.19E-6	--	--	1.26E-5
$T_1^{MQH_1 D_1}$	26	21	9.81E-8	ϕ	4.69E-8	1.30E-9	ϕ	1.43E-6
$T_2^{MQH_1 H'_2}$	27	1	4.71E-6	3.33E-6	1.76E-7	5.02E-8	5.12E-8	ϕ
$T_2^{MQD_1 D_2}$	29	3a	2.76E-8	ϕ	1.66E-6	ϕ	ϕ	2.33E-6
$T_3^{MQH_1 H'_2}$	33	1	2.02E-6	1.43E-6	7.54E-8	2.15E-8	2.20E-8	ϕ
$T_3^{QH_1 H'_2}$	39	1	3.33E-5	2.36E-5	1.24E-6	3.55E-7	3.62E-7	ϕ
$T_3^{QD_1 D_2}$	41	3a	1.89E-7	ϕ	1.14E-5	ϕ	ϕ	1.60E-5
T_4^{MLE} *	51	38	4.65E-6	--	1.96E-6	--	--	6.72E-6
T_5^{MLE} *	59	46	3.09E-6	--	1.30E-6	--	--	5.35E-6

Table B.27 Total Accident Sequence Probabilities Including Containment
Systems Failures With Recovery Considered - Base Case

Core Melt Accident Sequence	Analysis Number	AND Boolean Sequence Number	Containment System Sequences					
			$\bar{Z}_L \bar{C} \bar{F}$	$\bar{Z}_L \bar{C} F$	$\bar{Z}_L C$	$Z_L \bar{C} \bar{F}$	$Z_L \bar{C} F$	$Z_L C$
			* $\bar{Z}_T \bar{C}$		$\bar{Z}_T C$			Z_T
$S_3^{MH_1 H'_2}$	8	9	1.52E-5	8.09E-6	4.88E-7	2.35E-8	1.71E-8	
$S_3^{MED_2}$	10	11			5.80E-6			8.05E-7
S_3^{MXE}	11	12	2.12E-6					5.55E-8
T_1^{MLE} *	15	22	1.04E-6		2.38E-7			2.02E-5
T_2^{MLE} *	17	30	1.51E-6		1.10E-7			7.81E-7
T_3^{MLE} *	19	30a	7.19E-7		5.24E-8			3.63E-7
$T_1^{MQLD_1}$	26	21						7.48E-7
$T_2^{MQH_1 H'_2}$	27	1	1.19E-6	5.72E-7	2.42E-8	1.66E-9	1.13E-9	
$T_2^{MQD_1 D_2}$	29	3a			5.65E-7			1.41E-7
$T_3^{MQH_1 H'_2}$	33	1	5.10E-7	2.45E-7	1.04E-8	7.09E-10	4.85E-10	
$T_3^{QH_1 H'_2}$	39	1	8.41E-6	4.04E-6	1.71E-7	1.17E-8	8.01E-9	
$T_3^{QD_1 D_2}$	41	3a			3.87E-6			9.70E-7
T_4^{MLE} *	51	38	3.17E-6		7.42E-7			2.28E-7
T_5^{MLE} *	59	46	1.75E-6		4.47E-7			3.00E-7
			3.57E-5	1.30E-5	1.25E-5	3.76E-8	2.67E-8	2.47E-5

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6.0 VALUE ANALYSIS

6.1 Core Melt Probabilities

The purpose of this section is to determine and present the estimated core melt probabilities for the alternatives selected to address the potential vulnerabilities identified in Section 5.4 of the appendix. Further discussion of the modifications proposed for these vulnerabilities is given in Section 2.3 of the main report.

In the value analysis it is the change in core melt probability that results from an alternative that is most important. Nevertheless, it is informative to note the anticipated change in core melt probability due to each individual modification. This is done in Section 6.1.1.

6.1.1 Modification Values

The core melt probabilities for each of the internal vulnerabilities after the proposed modifications have been implemented will be estimated and discussed in this section.

Internal Vulnerability 1 - Failure of the Emergency Feedwater System turbine driven pump along with other safety system failures. The common factor in this vulnerability is the turbine driven pump where the motor driven pump has failed or power to it has failed, and feed and bleed has failed. The proposed modification is to use the existing Auxiliary Feedwater motor driven pump as a backup to the turbine driven pump. This involves connecting the Auxiliary Feedwater pump to class 1E power and installing a direct line to the CST. Credit for this pump is then determined by failure to actuate or failure of the pump or associated motor operated valve. These probabilities are estimated as follows:

PMD-LF (STANDBY)	8.7E-3
PMD-UTM	2.8E-4
VCC-LF	8.0E-3
VCC-UTM	3.0E-5
MANACT	<u>1.0E-3</u>

1.8E-2

The main feedwater flow paths are assumed to be open and the cross-tie operable if necessary.

A more accurate estimate considers the components in the two main feedwater paths. Failure of the three parallel valves in each line can be ignored. This leaves the following model:

FAILURE OF MAIN FEEDWATER PATHS TO THE STEAM GENERATORS

= p(MFW paths)

$$(1B+2B+3B+HB) * (1A+2A+3A+HA+8A)$$

where

$$1A\&1B = p(COC-LF) = 3.6E-5$$

$$2A\&2B = p(VOC-LF) = 3.6E-5$$

$$3A\&3B = p(XOC-LF) = 1E-4$$

$$HA\&HB = p(HTX-LFB) = 3E-4$$

$$8A = p(VCC-LF) = 8E-3$$

and UTM is not considered possible since the plant would be shutdown

Thus, $p(\text{MFW paths}) = (4.72E-4) * (1.27E-3) = 5.99E-7$ which is negligible with respect to the $1.8E-2$.

Thus, the estimate of core melt probability for vulnerability 1 given the modification is applied is:

$$6.77E-5 * 1.8E-2 = 1.2E-6$$

If this were to be considered as a recovery action, the recovery factor would be >0.1 which is consistent with the estimate.

Internal Vulnerability 2 - Common mode failure of valves in various safety systems. No modification is proposed.

Internal Vulnerability 3 - Common mode failure of safety system pumps. No modification is proposed.

Internal Vulnerability 4 - Diesel generator local fault and common mode failures with failure of the Emergency Feedwater turbine driven pump. This vulnerability involves two basic causes and overlaps with vulnerability 1. The suggested modification for the diesel generator failure and battery failure in vulnerability 5 is a turbine driven generator. Estimates for the turbine driven generator (TDG) can be derived from local faults of turbine driven pumps and diesel driven generators. The applicable factor for this modification is:

TDG-LF	estimate	3.0E-2
TDG-UTM	conservative estimate	6.0E-3
Failure to close circuit breaker		3.0E-3
open circuit of cables		
3E-6/hr*24 hr/day*30 days =		2.2E-3
MANACT		<u>1.0E-3</u>

$$4.2E-2$$

It is more accurate to consider also the steam supply to the turbine which comes from the EFWS TDP steam line. This is:

STEAM SUPPLY FAILURE (SSF) =
 (CV2666 + Q6)(CV2617 + Q5) + one XCC
 which is, SSF =
 $(VOC-LF)^2 + 2(VOC-LF)(VOC-UTM) + 2(COC-LF)(VOC-LF) +$
 $2(COC-LF)(VOC-UTM) + (COC-LF)^2 + XCC-LF + XCC-UTM$
 where
 $p(VOC-LF) = 3.6E-5$
 $p(VOC-UTM) = 1.1E-4$ and a double UTM is not allowed
 $p(COC-LF) = 3.6E-5$
 $p(XCC-LF) = 1E-4$
 $p(XCC-UTM) = 3E-5$
 $p(COC-UTM) = 0$ and thus was ignored
 and
 $p(SSF) = (3.6E-5)^2 + 2(3.6E-5)(1.4E-4) +$
 $2(3.6E-5)(3.6E-5) + 2(3.6E-5)(1.1E-4) +$
 $(3.6E-5)^2 + 1E-4 + 3E-5 = 1.3E-4$

which is negligible with respect to the $4.2E-2$.

The estimate of core melt probability for vulnerability 4 given the addition of the turbine driven generator is:

$$5.25E-6 * 4.2E-2 = 2.2E-7$$

If modification 1 is also considered the estimate after both modifications becomes

$$5.25E-6 * 1.8E-2 * 4.2E-2 = 4.0E-9$$

Obviously this is more decrease in core melt probability than necessary but the modification is really needed for vulnerability 5.

Internal Vulnerability 5 - Battery local fault and common mode failures. Loss of both batteries results in loss of instrumentation and control including failure to start the diesel generators. The modification proposed has already been described under vulnerability 4. Thus, the requirement would be

$$1.56E-5 * 4.2E-2 = 6.6E-7$$

There is one other failure probability that could be included, i.e., the failure of the charger, but at $2.4E-5$ it is insignificant.

Internal Vulnerability 6 - Failure of the Low Pressure Injection pumps. Failure of both low pressure pumps or one pump and the associated valves, power, or cooling of the other train fail recirculation since the High Pressure Recirculation System requires the low pressure output for suction head from the sump. The modification involves a parallel train that can swing to either existing train and that can use either power train. Failure of the new train is estimated from the

components needed to connect it to either existing train:

PMD-LF	8.7E-3
PMD-UTM	2.8E-4
2 VCC-LF 2*8E-3	1.6E-2
2 VCC-UTM 2*3E-5	6.0E-5
CCC-LF	1.0E-4
MANACT	<u>1.0E-3</u>
	2.6E-2

The estimate of core melt probability for vulnerability 6 given the addition of a swing low pressure tank is:

$$1.06E-5 * 2.6E-2 = 2.8E-7$$

Internal Vulnerability 7 - Failure to manually initiate Feed and Bleed or high pressure recirculation. The operator error probability assigned to all manual operations was 3E-3. This is probably conservative so that an increased awareness of the importance of these operator actions along with increased emphasis in regular training should decrease the event probability by a factor of 3 to 1E-3. Therefore, there is no hardware modification proposed for this vulnerability.

Internal Vulnerability 8 - Failure of the single manual discharge valve from the BWST or the BWST discharge valves leading to the safety system trains. This vulnerability is driven by the valve failure rates such that an obvious modification is parallel valves to provide an alternate flow path given failure of an existing valve. The pneumatic operated valves would receive the same actuation signals as the valve it parallels. This involves two valve modifications as shown below:

Manual valve	XOC-LF	1.0E-4
	XOC-UTM	<u>1.1E-4</u>
		2.1E-4
Pneumatic operated valve	NCC-LF	8.0E-3
	NCC-UTM	<u>1.1E-4</u>
		8.1E-3

The result is that the core melt probability becomes:

$$\begin{aligned}
 4.18E-6 * 2.1E-4 &= 8.8E-10 \\
 \underline{1.74E-6} * 8.1E-3 &= \underline{1.4E-8} \\
 5.92E-6 \text{ (original)} &1.5E-8 \text{ (modified)}
 \end{aligned}$$

6.1.2 Alternative Values

Only one combination of internal modifications is proposed for ANO-1 and this includes modifications or improvements to vulnerabilities 1, 4, 5, 6, 7, and 8.

The core melt probability value of an alternative will be calculated in Section 7 of the main report. The input to that calculation from the internal analysis is the base line core melt probability determined in Section 5.2 and the core melt probability given the modifications in the alternatives have been implemented which is the subject of this section.

Base Core Melt Probability - The base core melt probability from Section 5.2 is $7.191\text{E-}5$ including the eight vulnerabilities and recovery. The residual core melt probability of $1.567\text{E-}5$ must be added to this to account for accident sequences and cut sets that were not analyzed in depth. The total basecase core melt probability is $8.758\text{E-}5$.

Alternative 1 Core Melt Probability - This alternative involves all the internal modifications. Table B.28 delineates the top 14 or dominant accident sequences after recovery is applied. The total core melt probability is $3.26\text{E-}5$ plus $1.72\text{E-}6$ for residuals, which equals $3.436\text{E-}5$.

The core melt probability and containment system failure probabilities for alternative 1 were estimated using the alternative factors (AF) shown below rather than going to the expense of changing the models to reflect the modifications and going through the entire analysis again. We believe the end result is within a factor of 2 of that which would have been obtained by a complete analysis.

AF-1 = $1.8\text{E-}2$ *	EFWOP7AX-PTD-LF,UTM and other EPW- events	AFWS backup
AF-2 = $4.2\text{E-}2$ *	DG-CM DG1-GEN-LF,UTM DG2-GEN-LF,UTM BATT-CM BATTD06-ZBT-LF,UTM BATTD07-ZBT-LF,UTM	TDG backup
AF-3 = $2.6\text{E-}2$ *	LPIOP34A-PMD-LF,UTM LPIOP34B-PMD-LF,UTM	Parallel PMD
AF-4 = $3.3\text{E-}1$ *	PNB-MANACT HPRS-MANACT	Operator awareness
AF-5 = $2.1\text{E-}4$ *	LP1BWIX-XOC-LF,UTM	Parallel XOC

Table B.26 ANO-1 Results After Recovery -
Top 14 Sequences Alternative 1

Analysis Number	Core Melt Failure Sequence	Core Melt Probability	Boolean Sequence Number
8	$S_3MH_1H'_2$	1.43E-5	9
15	T_1MLE	7.15E-7	22
39	$T_3QH_1H'_2$	7.68E-6	1
10	S_3MED_2	3.82E-6	11
41	$T_3QD_1D_2$	2.87E-6	3a
51	T_4MLE	6.61E-8	38
59	T_5MLE	2.27E-7	46
17	T_2MLE	3.41E-8	30
11	S_3MXE	8.65E-7	12
27	$T_2MQH_1H'_2$	1.09E-6	1
19	T_3MLE	1.63E-8	30a
26	T_1MQLD_1	3.26E-8	21
33	$T_3MQH_1H'_2$	4.66E-7	1
29	$T_2MQD_1D_2$	4.19E-7	3a
		3.264E-5	

AF-6 = $8.1E-3$ * LPI1407A-VCC-LF,UTM
LPI1408B-VCC-LF,UTM

Parallel
NCC

There are other components that could be affected (i.e., EFWS-PMD, various valves in series with the pump) but although their contributions should not be very significant, some were considered in the application of the alternative factors.

Alternative 2 Core Melt Probability - This alternative core melt probability is derived from the basecase Table B.12 where the sequences without offsite power are multiplied by $1.0E-1$ and those with offsite power available are multiplied by $5.2E-2$ to account for the improvement obtained by the add-on system (reference Section 4.0 in the main report). Similarly, the residual must be apportioned regarding the availability of offsite power. This results in the new residual being $(6.58E-7)*(1.0E-1) + (3.73E-6)*(5.2E-2) = 2.60E-7$. The alternative 2 core melt probability is the sum of the new residual and the adjusted core melt probability of the top 14 accident sequences found in Table B.29 ($5.38E-6$) which equals $5.644E-6$.

Only the summary tables are presented in this section although all of the accident sequences were run through the SETS program and tabulated to obtain the applicable residual core melt probability.

6.2 Containment System Probabilities

The containment system probabilities were obtained applying the same multiplier factors used to obtain the core melt probabilities. The alternative factors (AF) were applied to each cut set of the top accident sequences for alternative 1 and the residuals reduced accordingly. The add-on factors with and without offsite power were applied to each containment system sequence for alternative 2. The end result is Table B.30 and B.31.

Table B.29 ANO-1 Results After Recovery -
Top 14 Sequences Alternative 2

Analysis Number	Core Melt Failure Sequence	Core Melt Probability	Boolean Sequence Number
8	$S_3MH_1H'_2$	1.16E-6	9
15	T_1MLE	2.13E-6	22
39	$T_3QH_1H'_2$	6.29E-7	1
10	S_3MED_2	3.43E-7	11
41	$T_3QD_1D_2$	2.38E-7	3a
51	T_4MLE	2.12E-7	38
59	T_5MLE	1.28E-7	46
17	T_2MLE	1.24E-7	30
11	S_3MXE	1.21E-7	12
27	$T_2MQH_1H'_2$	8.89E-8	1
19	T_3MLE	5.88E-8	30a
26	T_1MQLD_1	7.84E-8	21
33	$T_3MQH_1H'_2$	3.81E-8	1
29	$T_2MQD_1D_2$	3.46E-8	3a
		5.384E-6	

Table B.30 Total Accident Sequence Probabilities Including Containment
Systems Failures With Recovery Considered - Alternative 1

Core Melt Accident Sequence	Analysis Number	ANO Boolean Sequence Number	Containment System Sequences					
			$\bar{Z}_L \bar{C}\bar{F}$	$\bar{Z}_L \bar{C}F$	$\bar{Z}_L C$	$Z_L \bar{C}\bar{F}$	$Z_L \bar{C}F$	$Z_L C$
			* $\bar{Z}_T^C$		$\bar{Z}_T^C$			Z_T
$S_3^{MH_1 H'_2}$	8	9	7.42E-6	7.75E-6	3.24E-7	8.65E-9	1.62E-8	
$S_3^{MED_2}$	10	11			2.97E-6			1.52E-6
S_3^{MXE}	11	12	7.51E-7					5.34E-8
T_1^{MLE} *	15	22	2.46E-9		7.11E-11			6.49E-7
T_2^{MLE} *	17	30	1.58E-8		8.44E-10			3.76E-8
T_3^{MLE} *	19	30a	7.51E-9		4.02E-10			8.32E-9
$T_1^{MQLD_1}$	26	21						3.29E-8
$T_2^{MQH_1 H'_2}$	27	1	5.83E-7	5.66E-7	1.33E-8	6.05E-10	1.13E-9	
$T_2^{MQD_1 D_2}$	29	3a			3.15E-7			1.34E-7
$T_3^{MQH_1 H'_2}$	33	1	2.50E-7	2.43E-7	5.68E-9	2.59E-10	4.85E-10	
$T_3^{QH_1 H'_2}$	39	1	4.12E-6	4.00E-6	9.38E-8	4.28E-9	8.01E-9	
$T_3^{QD_1 D_2}$	41	3a			2.16E-6			9.20E-7
T_4^{MLE} *	51	38	5.09E-8		1.25E-8			7.70E-9
T_5^{MLE} *	59	46	3.56E-8		1.28E-9			1.73E-7
			1.32E-5	1.26E-5	5.90E-6	9.94E-9	2.58E-8	3.54E-6

Table B.31 Total Accident Sequence Probabilities Including Containment
Systems Failures With Recovery Considered - Alternative 2

Core Melt Accident Sequence	Analysis Number	ANO Boolean Sequence Number	Containment System Sequences					
			$\bar{Z}_L \bar{C}\bar{F}$	$\bar{Z}_L \bar{C}F$	$\bar{Z}_L C$	$Z_L \bar{C}\bar{F}$	$Z_L \bar{C}F$	$Z_L C$
			* $\bar{Z}_T \bar{C}$		$\bar{Z}_T C$			Z_T
$S_3^{MH_1 H'_2}$	8	9	7.90E-7	4.21E-7	2.54E-8	1.22E-9	8.89E-10	
$S_3^{MED_2}$	10	11			3.02E-7			4.19E-8
S_3^{MXE}	11	12	1.10E-7					2.89E-9
T_1^{MLE} *	15	22	1.04E-7		2.38E-8			2.02E-6
T_2^{MLE} *	17	30	7.85E-8		5.72E-9			4.06E-8
T_3^{MLE} *	19	30a	3.74E-8		2.73E-9			2.11E-8
$T_1^{MQLD_1}$	26	21						7.48E-8
$T_2^{MQH_1 H'_2}$	27	1	6.19E-8	2.97E-8	1.26E-9	8.63E-11	5.88E-11	
$T_2^{MQD_1 D_2}$	29	3a			2.94E-8			7.33E-9
$T_3^{MQH_1 H'_2}$	33	1	2.65E-8	1.27E-8	5.40E-10	3.69E-11	1.52E-11	
$T_3^{QH_1 H'_2}$	39	1	4.37E-7	2.10E-7	8.89E-9	6.08E-10	4.17E-10	
$T_3^{QD_1 D_2}$	41	3a			2.01E-7			5.04E-8
T_4^{MLE} *	51	38	1.65E-7		3.86E-8			1.89E-8
T_5^{MLE} *	59	46	9.10E-8		2.32E-8			1.56E-8
			1.90E-6	6.73E-7	6.63E-7	1.95E-9	1.39E-9	2.29E-6

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APPENDIX C

Seismic Risk Assessment
of the
Arkansas Nuclear One - Unit One Power Plant

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1.0 INTRODUCTION

The US Nuclear Regulatory Commission has designated decay heat removal an Unresolved Safety Issue and established a program under Task Action Plan (TAP) A-45, Decay Heat Removal Requirements (Reference 1), to provide the technical basis for resolution of the issue. As part of the program to establish a technical base for decision making, probabilistic risk assessments are being performed for a number of example plants. These risk assessments are focused on the plant's response to small loss of coolant accidents (SLOCAs) and transient events which require the various decay heat removal systems to bring the plant to a cold shutdown state. As part of this risk assessment program, simplified seismic risk assessments are being performed for the plants under study. This report describes the seismic risk assessment for the Arkansas Nuclear One Power Plant.

In preparation for this work, a set of procedures for performing a simplified seismic risk assessment was developed (Reference 2). The result was a methodology that can be applied to a particular plant in a 16-week timeframe and at a cost of about \$40K per plant. The simplifications utilized in this methodology were derived from the extensive results obtained in the NRC-sponsored Seismic Safety Margins Research Program (Reference 3). The simplified methodology is summarized below.

1.1 Summary of Seismic Risk Analysis Procedures

There are seven steps required for calculating the seismic risk at a nuclear power plant:

1. Determine the local earthquake hazard (Hazard Curve and Site Spectra).
2. Identify accident scenarios for the plant which lead to radioactive release (Initiating Events and Event Trees).
3. Determine failure modes for the plant safety and support systems (Fault Trees).
4. Determine fragilities (probabilistic failure criteria) for the important structures and components.
5. Determine the responses (accelerations or forces) of all structures and components (for each earthquake level).
6. Compute the probability of core melt using the information from Steps 1 through 5.
7. Estimate uncertainty in the core melt frequencies.

Only the level of detail differentiates a simplified seismic analysis from a detailed seismic PRA. The seven steps of the TAP A-45 simplified seismic risk analysis procedure are described below.

Step 1 - Seismic Hazard Characterization

- (a) If a seismic hazard curve exists for the plant (e.g., from the Systematic Evaluation Program (SEP) or from an existing PRA of a nearby plant), it is used.
- (b) If no such curve exists, a hazard curve scaled to an exceedance probability of $2.5E-4$ /year at the safety shutdown earthquake (SSE) would be used. The slope of the curve for higher peak ground acceleration (PGA) values is estimated from hazard curves for other sites in the same broad seismological province.
- (c) All plants west of the Rocky Mountains require site specific hazard curves due to the high levels of seismic activity, but most existing plants have hazard curves already available.

Step 2 - Initiating Events and Event Trees

The scope of TAP A-45 includes only the initiating events associated with small and small-small loss of coolant accident (SLOCA) and transient events. Two types of transients are being considered: those in which the power conversion system (PCS) is initially available (denoted Type T3 transients) and those in which the PCS is failed as a direct consequence of the initiating event (denoted Type T2 transients). The event trees derived for the internal event analyses for TAP A-45 are utilized.

The small and small-small LOCA initiating event frequencies are computed based on a statistical distribution of small pipe failures developed in the NRC-sponsored Seismic Safety Margins Research Program (SSMRP) (Reference 3).

The frequency of Type T2 transients is based on the probability of loss of offsite power (LOSP). This is the dominant cause of this type of transient (for the majority of plants for which LOSP causes loss of main feedwater). The frequency of the Type T3 initiating event is computed from the condition that the sum of the initiating event probabilities must be unity. The hypothesis is that, given an earthquake of reasonable size, at least one of the initiating events will occur.

Step 3 - Fault Trees

Simplified system fault trees for a number of example plants have been developed as part of TAP A-45, (for random failures

only). These fault trees will be used, although they require some modification to include basic events for seismic failure modes and re-solving the trees for pertinent cutsets to be included in the probabilistic calculations. Probabilistic culling is used in re-solving these trees to assure that important correlated failure modes are not lost.

Step 4 - Component and Structure Failure Descriptions

Component seismic fragilities are obtained both from a generic fragility data base and from plant-specific fragilities developed for components identified during the plant walk-down.

The generic data base of fragility functions for seismically-induced failures was originally developed as part of the SSMRP (Reference 4). Fragility functions for the generic categories were developed based on a combination of experimental data, design analysis reports, and an extensive expert opinion survey. The experimental data utilized in developing fragility curves were obtained from the results of component manufacturer's qualification tests, independent testing lab failure data and data obtained from the extensive US Corps of Engineers SAFEGUARD Sub-system Hardness Assurance Program. These data were statistically combined with the expert opinion survey data to produce fragility curves for each of the generic component categories.

Within the 16-week/plant time scale, structural fragility development is not possible, and is not being included in TAP A-45. This is appropriate since it is the design requirements for decay heat removal systems which are being evaluated, and not the structural design. However, the simplified seismic analysis approach being applied should prove widely useful in examining other generic and value/impact issues not directly concerned with adequacy of power plant structures.

Step 5 - Seismic Response of Structures and Components

Building and component seismic responses are estimated from peak ground accelerations at several probability intervals on the hazard curve. Three basic aspects of seismic response--best estimates, variability, and correlation--are generated. SSMRP Zion analysis results and simplified methods studies form the basis for assigning scaling, variability and correlation of responses.

In each case, SHAKE code (Reference 5) calculations are performed to assess the effect of the local soil column (if any) on the surface peak ground acceleration and soil structure interactions. This permits an evaluation of the effects of non-homogeneous underlying soil conditions which can strongly affect the building responses.

Two situations are commonly encountered in developing structural responses. First, for many early plants, only design floor spectra are available. For this situation, the results of a detailed soil-structure interaction investigation (Reference 6) made as part of the SSMRP may be used to scale the computed design floor slab accelerations to best estimate values including the effects of the soil, and to generate best estimate floor response spectra.

Second, one has the situation where fixed-base mass-spring or eigen-system descriptions are available for one or more buildings at the site. For this case, one can compute the floor slab accelerations using the CLASSI code (Reference 3). This code takes a fixed-base eigen-system model of the structures and input-specified frequency dependent soil impedances and computes the structural response (as well as variation in structural response is desired).

Variability in responses (floor and spectral accelerations) is assigned based on the SSMRP results. The recommended uncertainties (expressed as standard deviations of the logarithms of the responses, β) are shown below:

<u>Quantity</u>	<u>β random</u>
Peak Ground Acceleration	0.25
Floor Zero Period Acceleration	0.35
Floor Spectral Acceleration	0.45

Correlation between component failures is being included explicitly in TAP A-45. In computing the correlation between component failures (in order to quantify the cut sets) it is necessary to consider correlations both in the responses and in the fragilities of each component. Inasmuch as there are no data as yet on correlation between fragilities, the fragility correlations between like components are taken as zero and 1, and the possible effect of such correlation quantified in a sensitivity study. The correlation between responses is assigned according to a set of rules. These are applied to both BWR and PWR plants.

Step 6 - Probabilistic Failure and Core Melt Calculations

Given the input from the five steps above, the SETS code (Reference 7) is used to calculate the required output (probabilities of failure, core melt, etc.). Use of the SEISIM code for final computation of accident sequences permits proper inclusion of seismically-induced correlation between component failures.

Step 7 - Estimate Uncertainties

Uncertainties in core damage estimates are being based entirely on upper and lower confidence limits on the hazard curve.

Generic recommendations for such limits are given in Reference 2. This simplification is appropriate since previous seismic PRAs have shown the hazard curve uncertainty to contribute 70-90% of the total seismic risk uncertainty.

2.0 PLANT DESCRIPTION

The Arkansas Nuclear One Unit-1 (ANO-1) nuclear power plant is an 836 MWe pressurized water reactor (PWR) located on Lake Dardanelle near Russellville, Arkansas. Arkansas Power and Light Company owns and operates the facility. ANO-1 entered commercial operation on December 19, 1974.

The reactor vendor for ANO-1 is Babcock and Wilcox (B&W). The architectural engineer is the Bechtel Power Corporation. The design of the reactor coolant system is typical of other B&W plants currently in commercial operation; these are two once-through steam generators and four reactor coolant loops. Most of the major safety system designs are also fairly typical of other B&W plants.

The plant is built on a site where shallow clay and silty clay deposits overlie bedrock consisting of Pennsylvanian McAlester Formation shale. Hard, fine-grained sandstone of the Harthshorne formation underlie the shale and was encountered at a depth of 150 ft. The plant foundation was placed on bedrock, so this is considered a rock site. Based on the ANO-1 Final Safety Analysis Report (Reference 8), 0.20g is defined as the SSE.

3.0 DESCRIPTION OF PLANT SYSTEMS

The important plant systems are briefly discussed with the functional interactions which cause the dominant seismic failure modes highlighted.

High Pressure Injection/High Pressure Recirculation

The high pressure (HP) system is utilized during those LOCAs where the reactor coolant pressure remains high (i.e., above about 150 psig). Figures 3.1 through 3.3 are simplified schematics of the HP system.

The HP system is a two-train, three-pump system which injects water into the reactor pressure vessel via four injection headers (one for each cold leg of the RCS). The injection headers are cross-connected such that each pump has an open flow path to all four reactor coolant system (RCS) cold legs. Suction in the injection mode is from the BWST, while in the recirculation mode it is from the low pressure injection system. Pumps A and B are powered from AC/DC train A and pump C is powered from AC/DC train B.

Low Pressure Injection/Low Pressure Recirculation

The LPI/LPR system serves a number of functions during both accident conditions and normal operations. Along with providing decay heat removal and filling and draining of the fuel transfer canal during plant outages and refueling, the system also provides emergency core cooling (ECC) during a LOCA. Figures 3.4 through 3.8 are simplified schematics of the LPI/LPR system.

The LPI/LPR system consists of two independent trains which draw water via a common header from the BWST in the injection mode and the reactor building sump in the recirculation mode. Each train of the LP system consists of an LP pump, decay heat cooler, piping and valves. Water is injected directly into the reactor pressure vessel through the two cross-tied injection headers. LP pumps A and B are powered from AC/DC trains A and B respectively.

Emergency Feedwater System

The emergency feedwater (EFW) system is a backup for the main feedwater system (MFWS). Post-shutdown decay heat is removed from the reactor coolant system (RCS) via the steam generators (SGs). Figure 3.9 is a simplified schematic of the EFW system.

The EFW system consists of two interconnected trains, capable of supplying feedwater to either or both SGs. Train A contains a motor-driven pump, and the pump of train B is turbine driven. The system pumps take suction from the condensate storage tank (CST). The motor-driven pump is

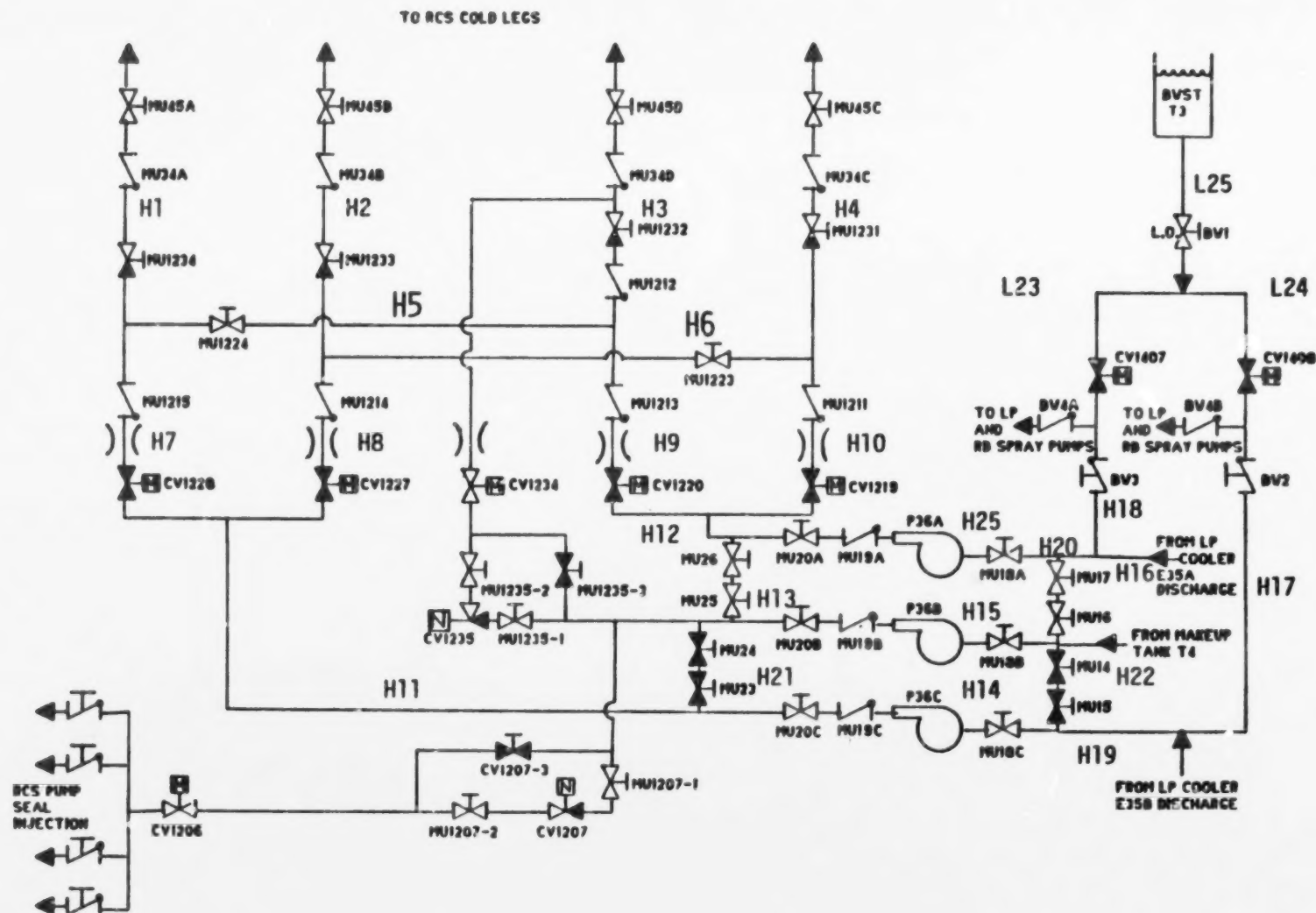


Figure 3-1 Pipe Segments of High Pressure Injection System

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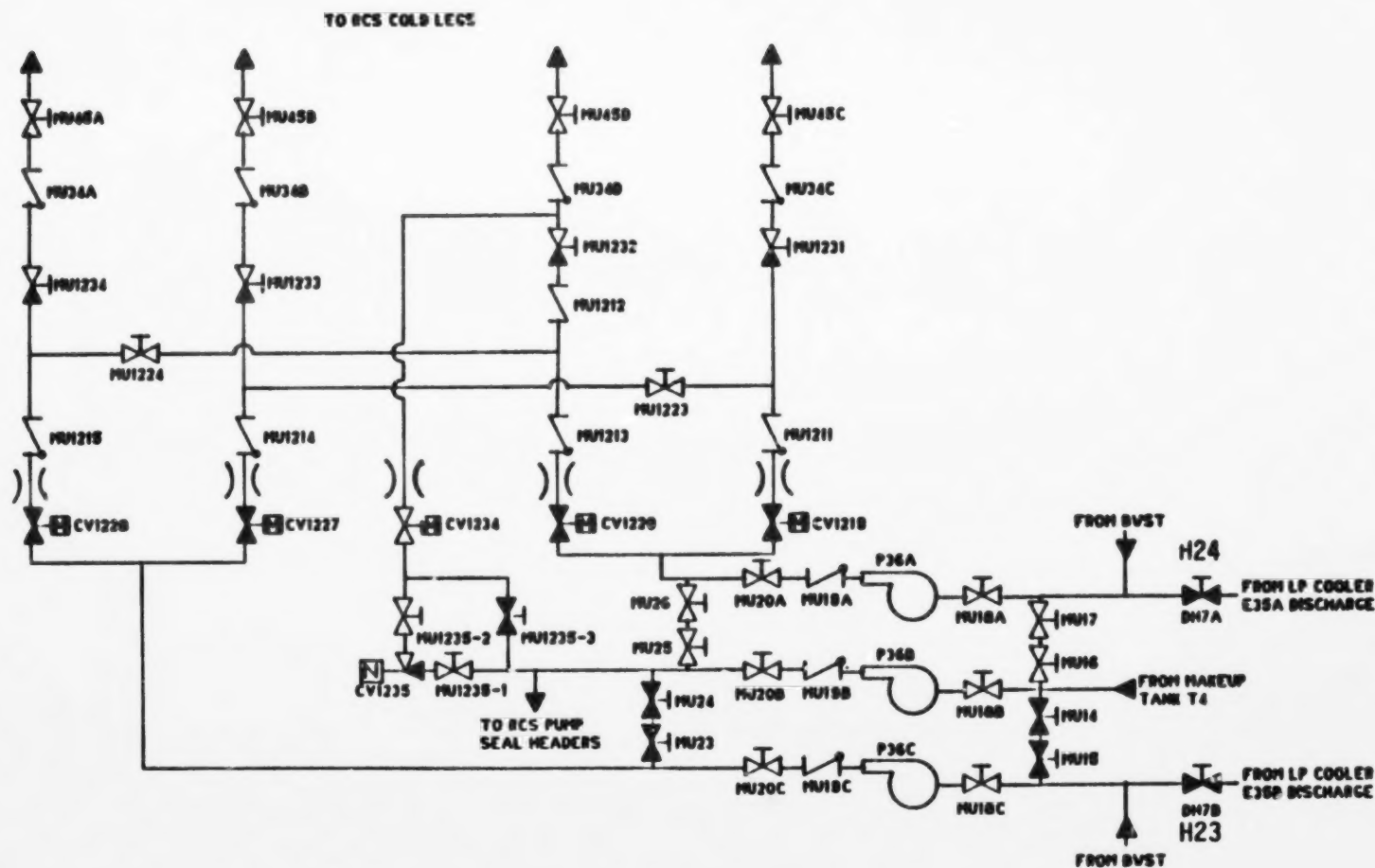


Figure 3-2 Pipe Segments for High Pressure Recirculation System

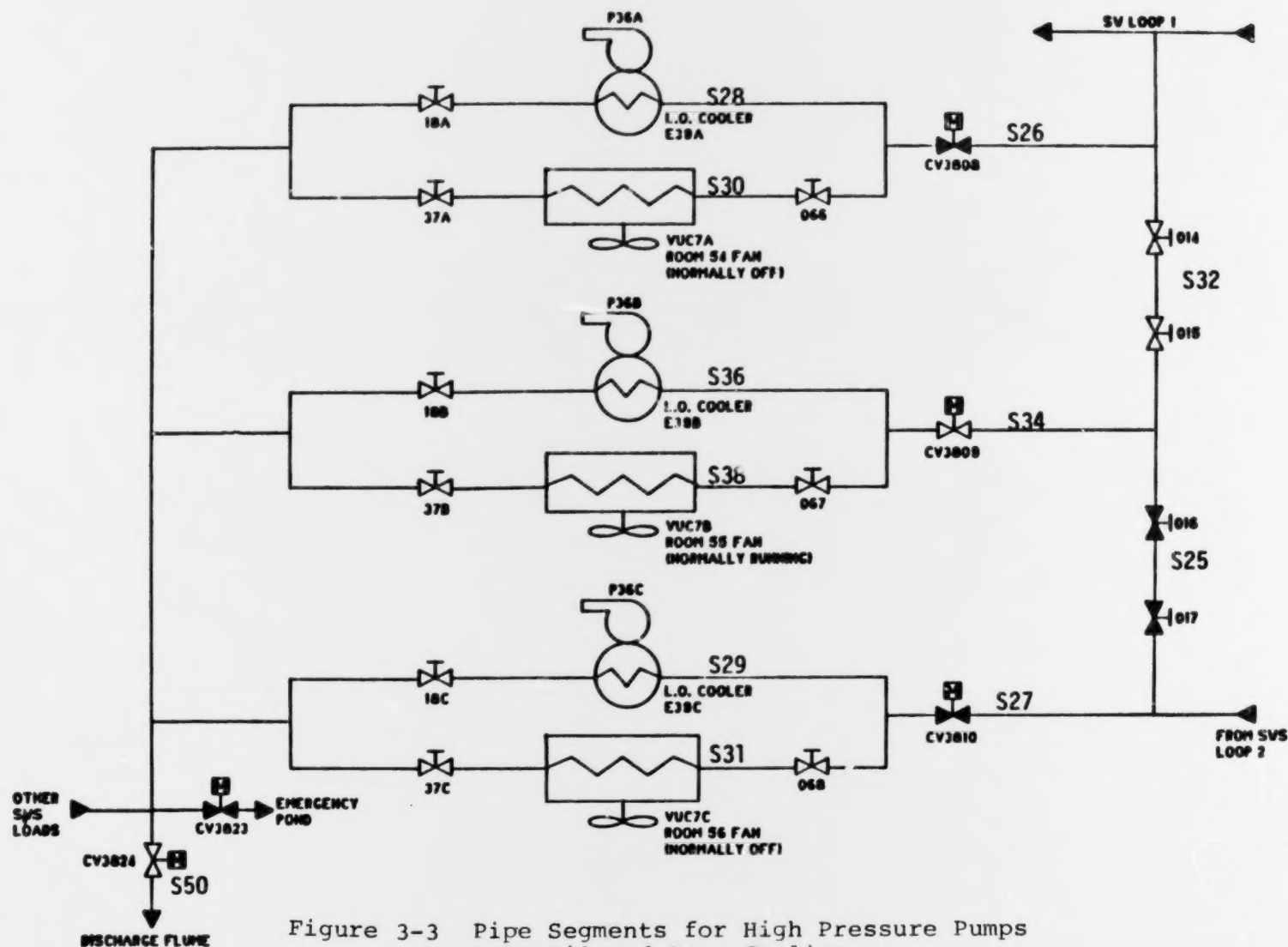
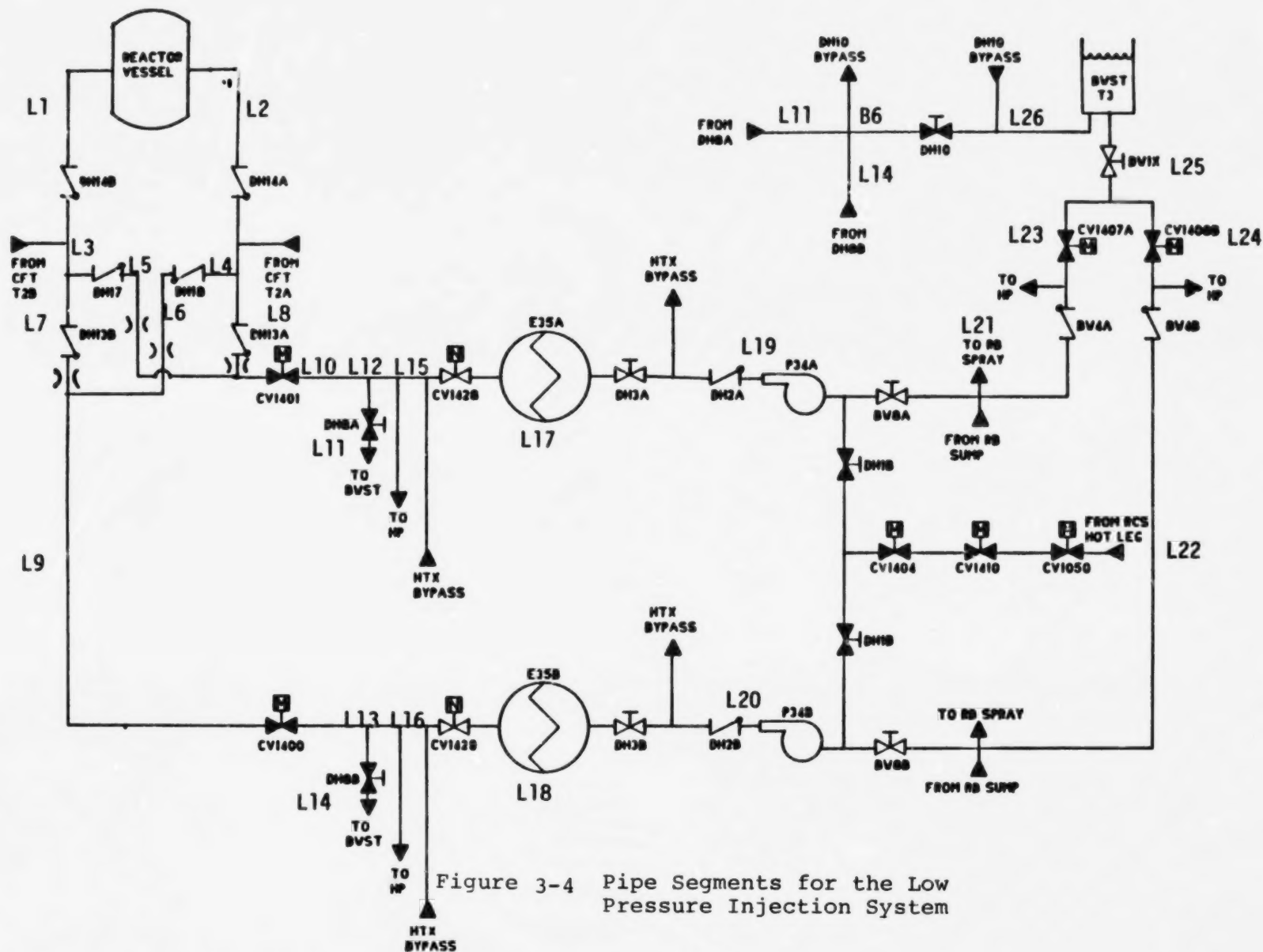


Figure 3-3 Pipe Segments for High Pressure Pumps
Lube Oil and Room Cooling

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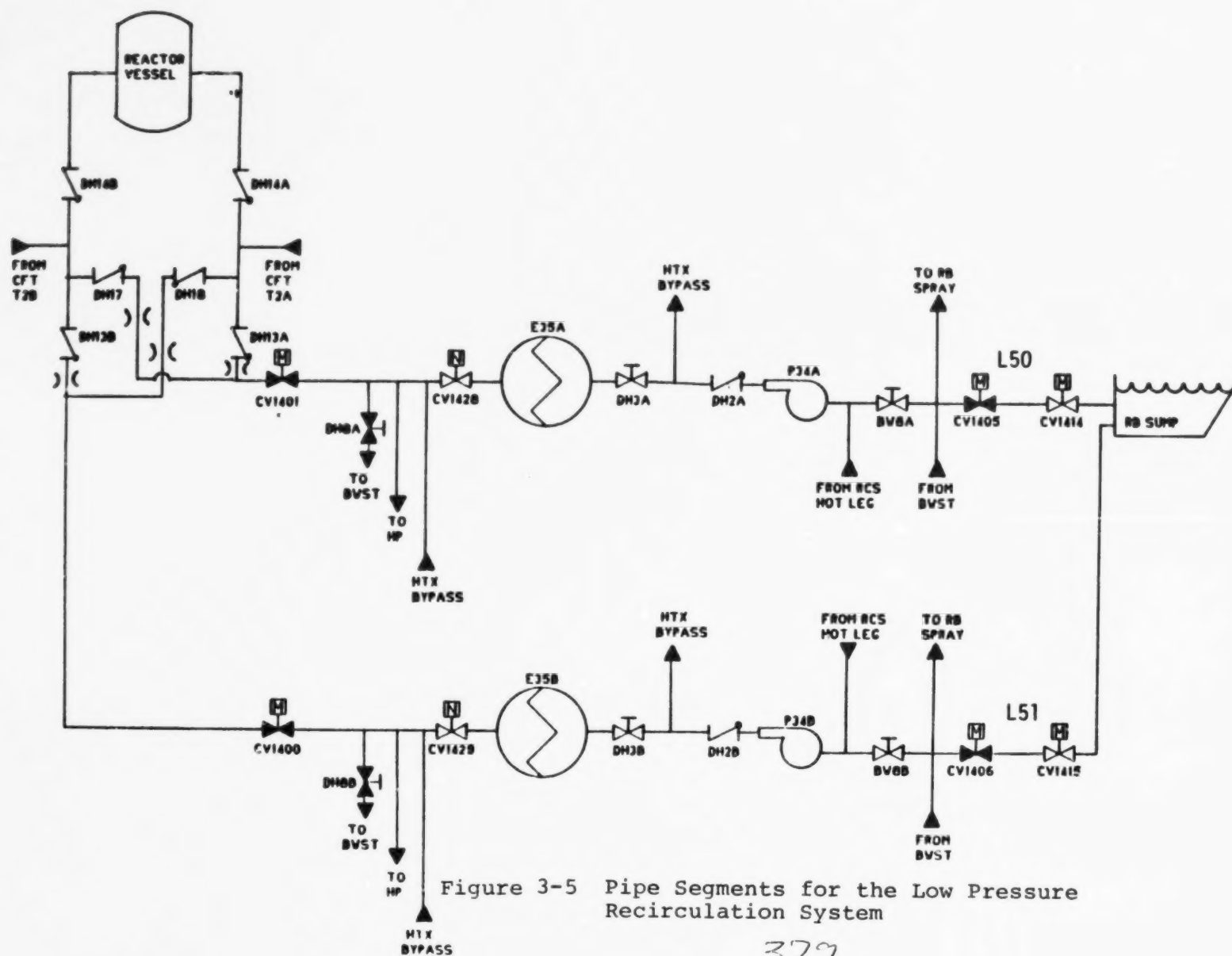


Figure 3-5 Pipe Segments for the Low Pressure Recirculation System

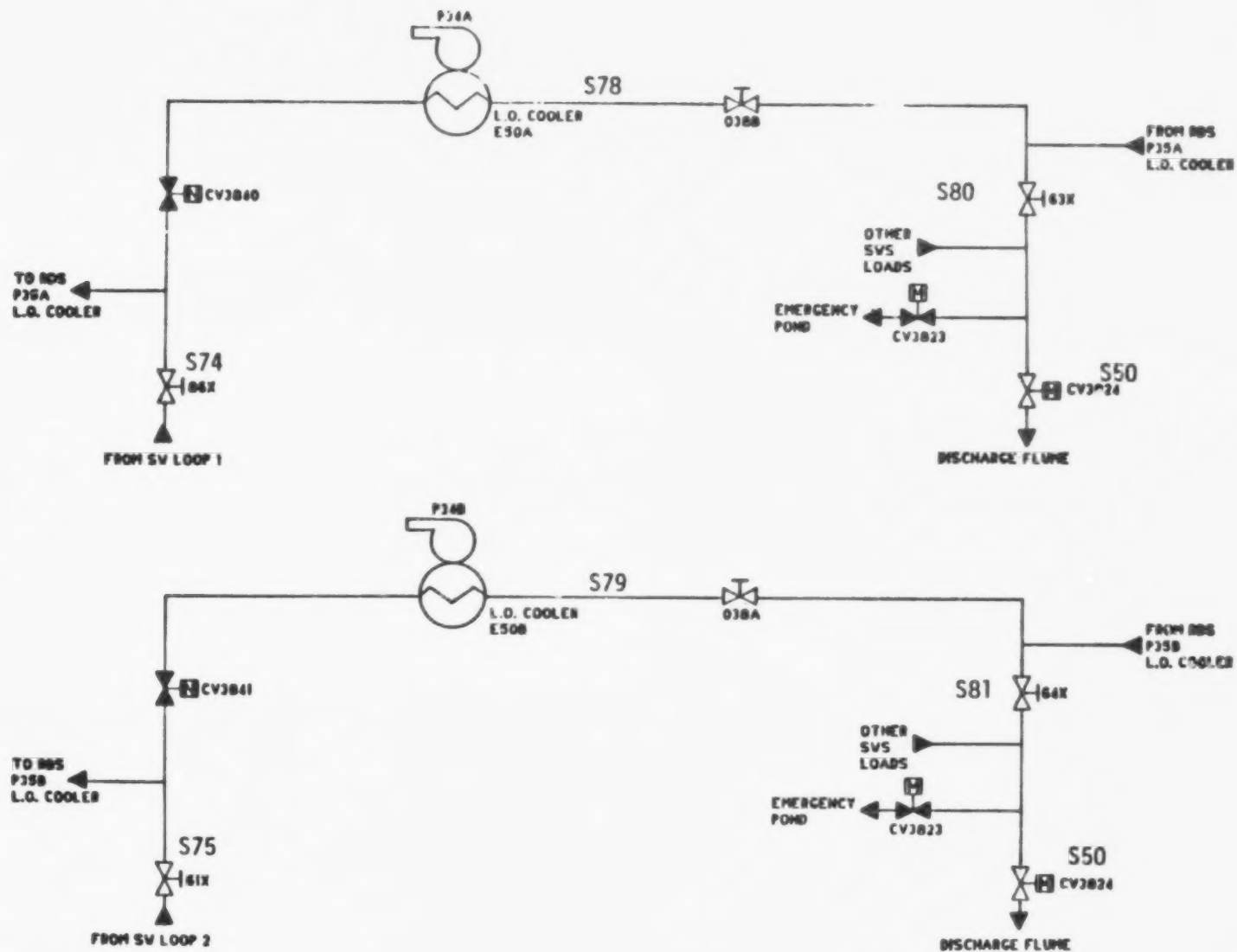


Figure 3-6 Pipe Segments for the Low Pressure Pump Lube Oil Coolers

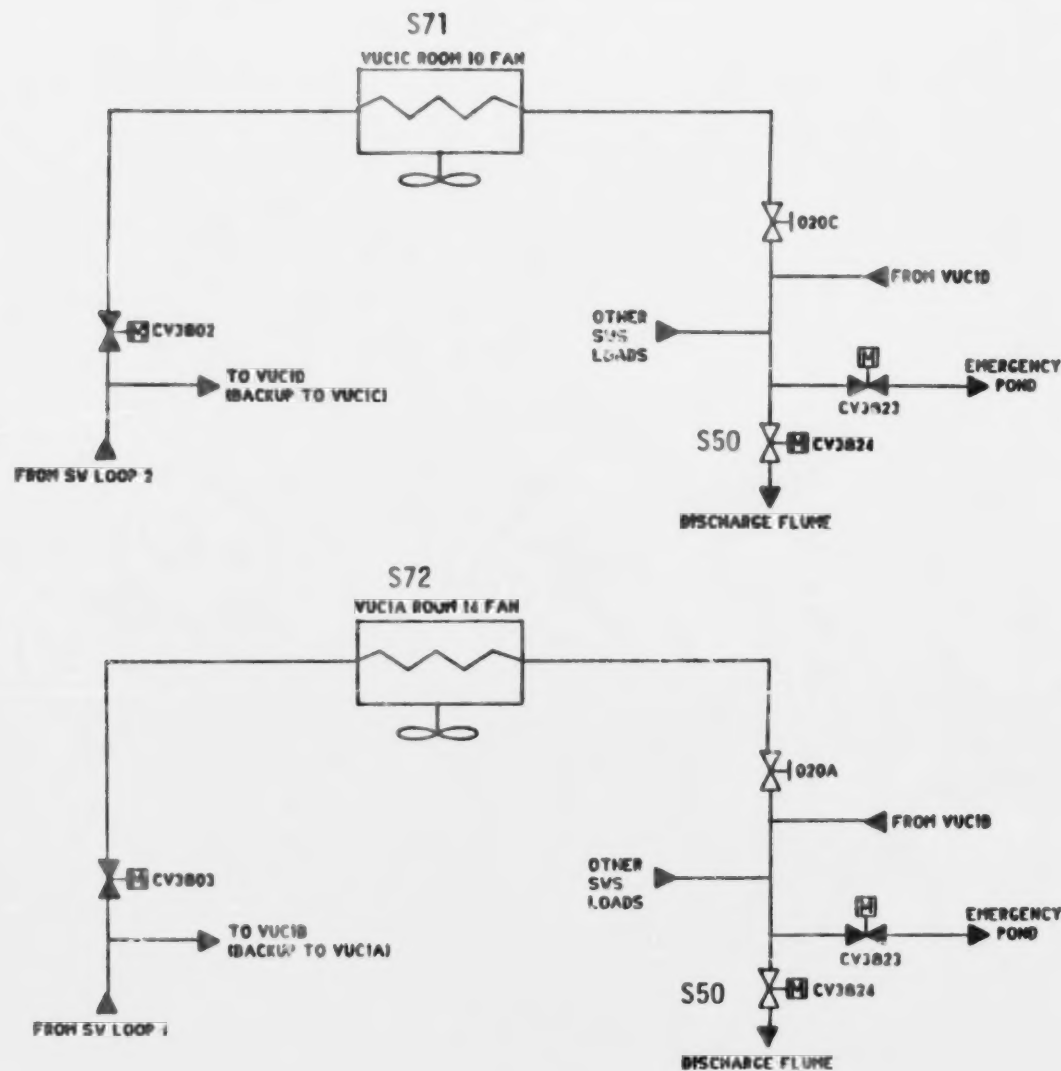


Figure 3-7 Pipe Segments for Low Pressure Pump Room Coolers

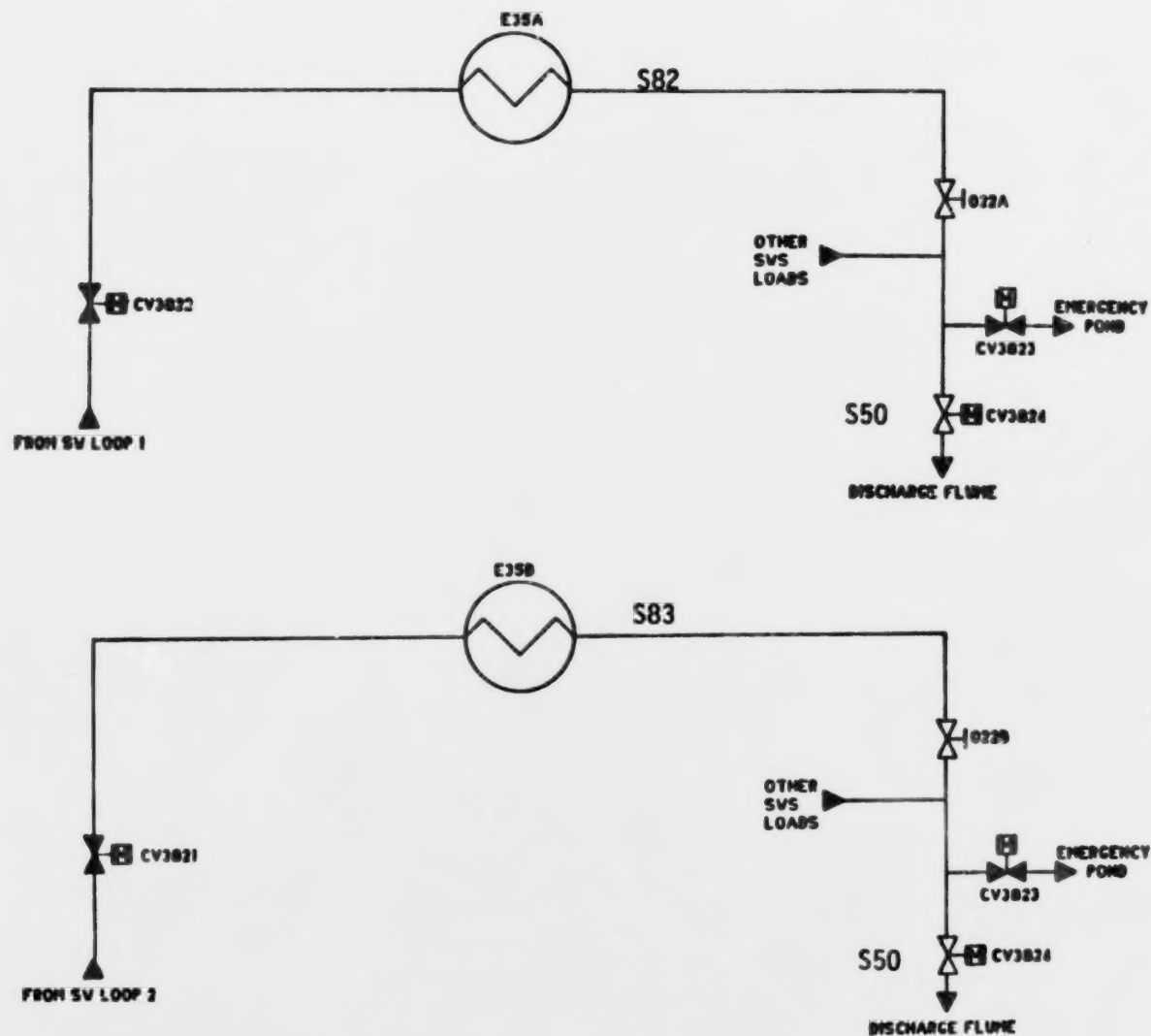


Figure 3-8 Pipe Segments for Low Pressure Decay Heat Exchangers Cooling

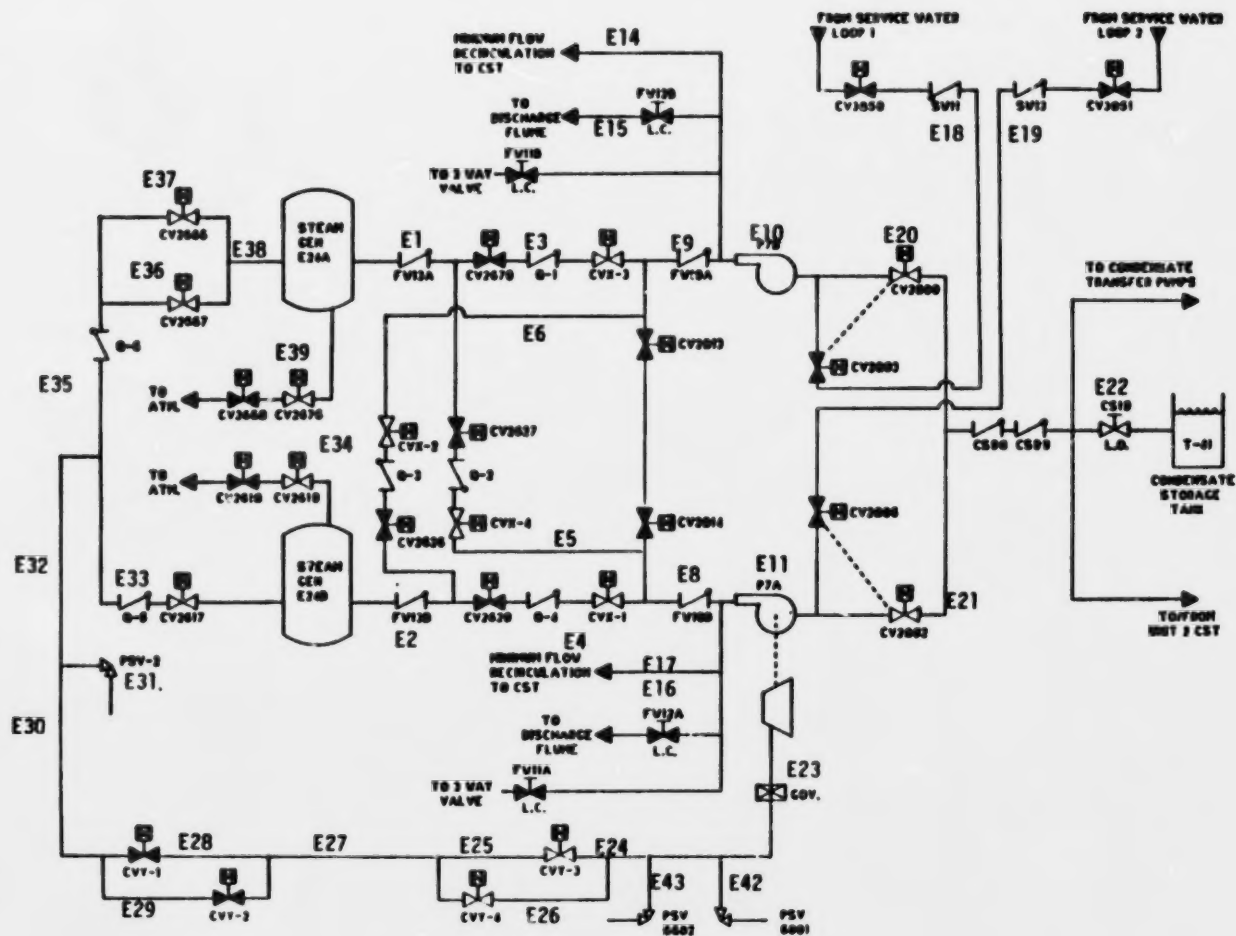


Figure 3-9 Pipe Segments for the Emergency Feedwater System

powered from AC/DC train A, while the steam admission valves A and B for the turbine-driven pump are powered from DC trains A and B respectively.

Reactor Building Cooling System

The reactor building cooling system (RBCS) limits post-accident reactor building pressure during steam evolution within the building due to an accident. Figure 3.10 is a simplified schematic of the RBCS.

The reactor building atmosphere enters each of the fan coolers at the fan locations. All four fans discharge into a supply air plenum which distributes cooled air throughout the reactor building.

Power Conversion System

The power conversion system (PCS) provides feedwater to the secondary side of the steam generators which, in turn, transfers energy to the turbine generator system. Figure 3.11 is a simplified schematic of the PCS.

The feedwater portion of the PCS consists of two pump trains. Three low pressure motor-driven condensate pumps feed two high pressure steam-driven main feedwater pumps and one high pressure motor-driven auxiliary feedwater pump. These latter three pumps in turn feed both steam generators via two injection lines.

Service Water System

The service water system (SWS) provides cooling water to the following equipment during emergency conditions: (1) reactor building cooling system cooling coils, (2) diesel generator jacket heat exchangers, (3) high pressure pump lube oil coolers, (4) high pressure pump room coolers, (5) circulating water pumps bearing lubrication, (6) low pressure/building spray pump room coolers, (7) low pressure pump(s) bearing coolers, (8) low pressure system heat exchangers, (9) building spray pump(s) bearing lube oil coolers, and (10) emergency feedwater system water sources. Figure 3.12 is a simplified schematic of the SWS.

The SWS consists of two redundant loops. Normal cooling is supplied from Lake Dardanelle with an emergency pond available as a backup. There are three SW pumps. During normal operation, two are running with one in standby.

Emergency AC Electrical System

The emergency AC electrical (EAC) system provides electrical power to ESF equipment.

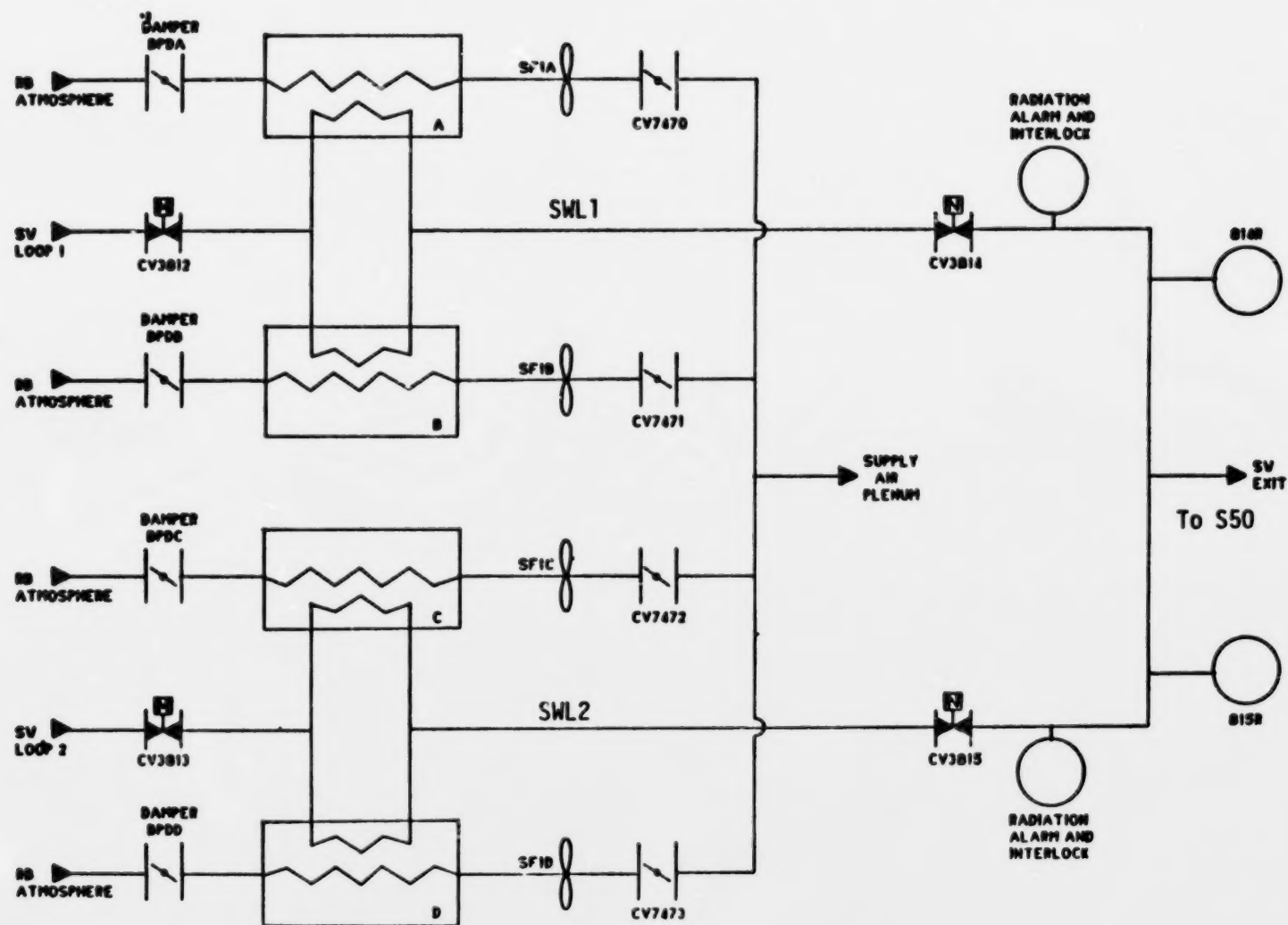


Figure 3-10 Pipe Segments for the Reactor Bldg. Cooling System

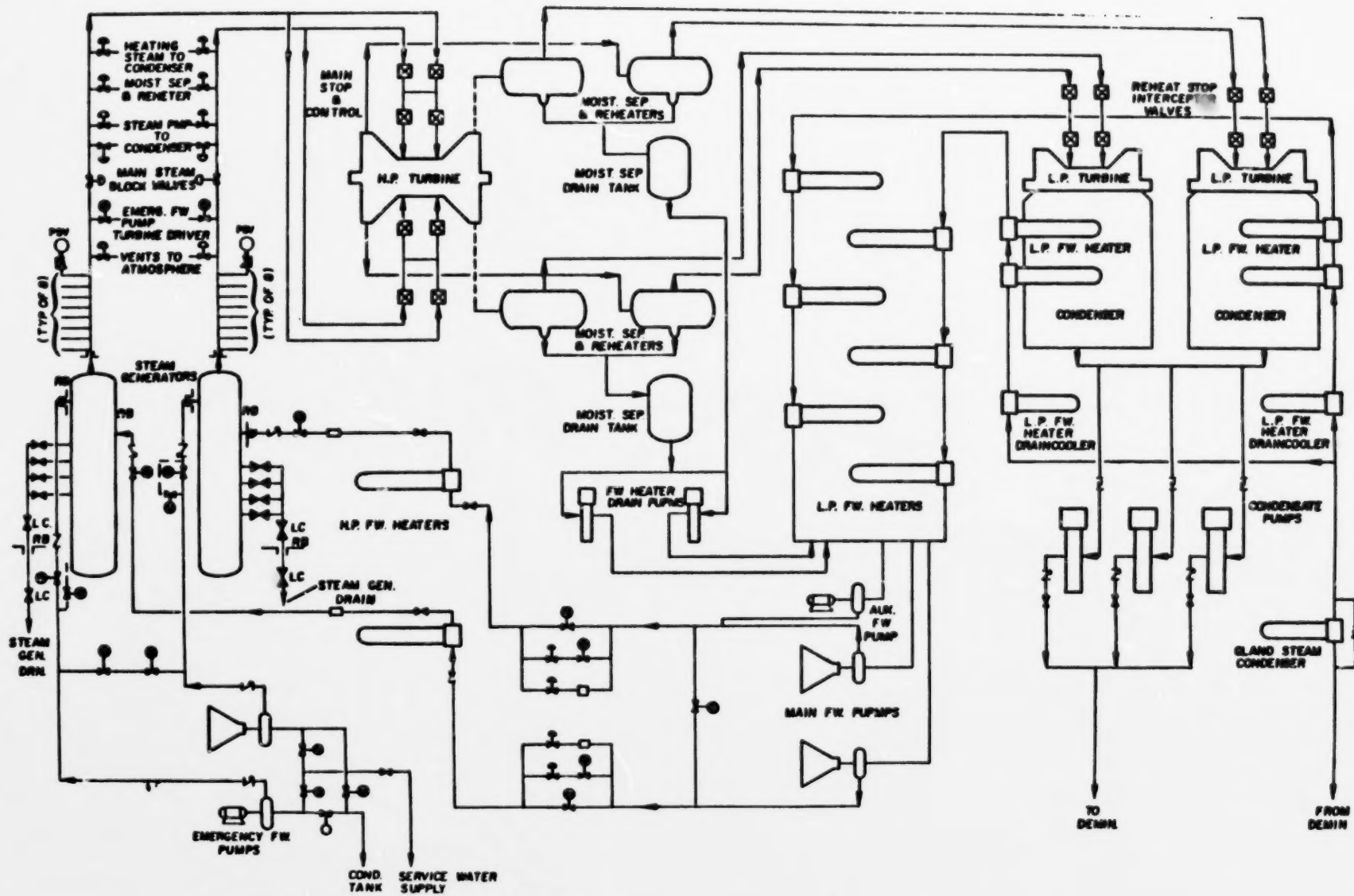


Figure 3-11 Schematic of Power Conversion System

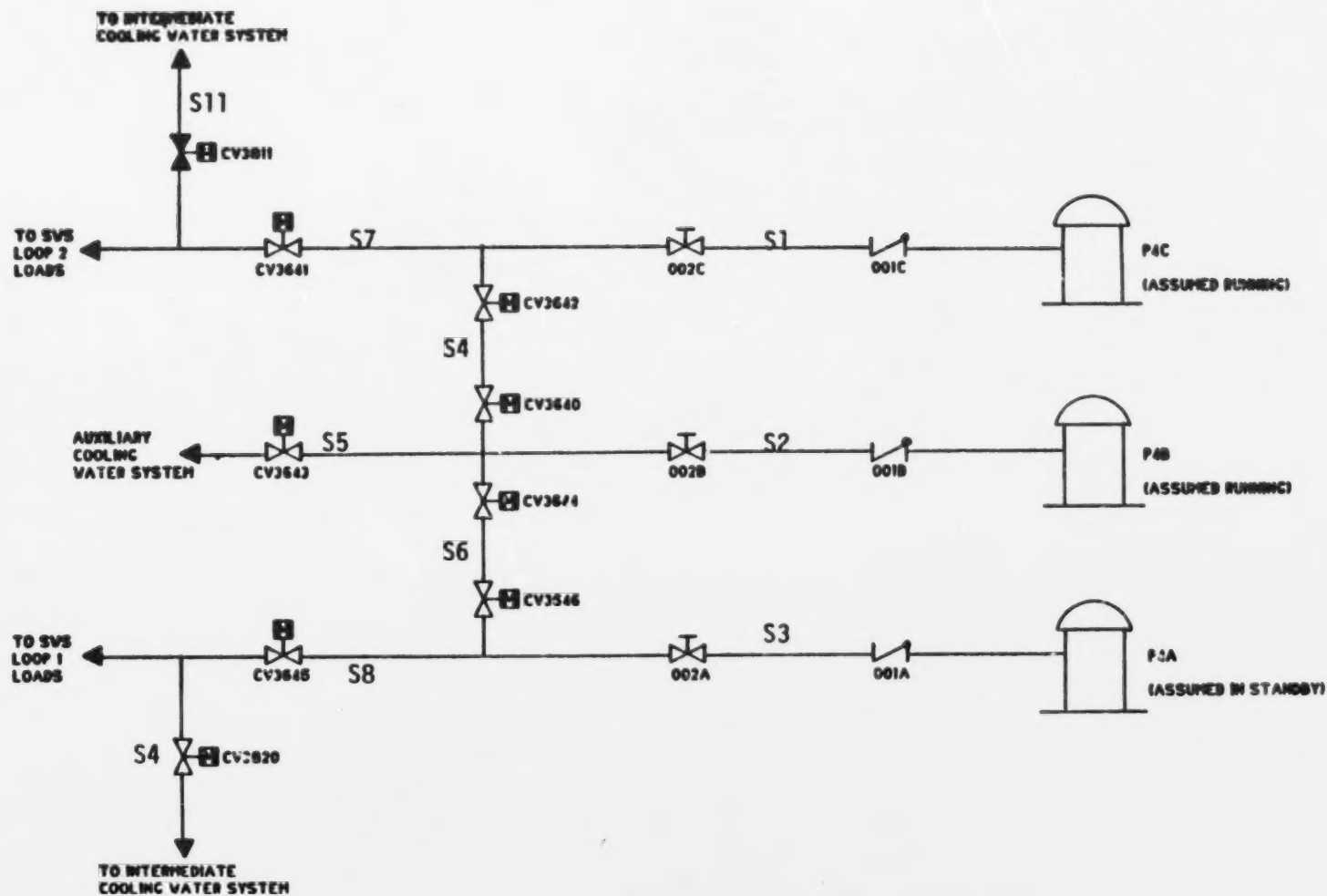


Figure 3-12 Pipe Segments for Service Water System Pump Loops

The EACS is composed of two trains, each comprising a diesel generator, 4160V switchgear, 480V load centers and motor control centers, 120V instrumentation panels, and associated transformers and circuit breakers. The normal power supply to the two trains is offsite.

As can be seen from Figure 3.13, the two trains are capable of being cross-tied. Normal practice at ANO-1 is not to cross-tie the two trains. Figure 3.14 shows the service water dependency of the diesel generators.

DC Power System

The 125-volt DC system provides continuous power for control, instrumentation, reactor protection and engineered safeguards actuation systems, and emergency safeguard actuation control systems. In addition, it powers the control valves in the emergency feedwater system and provides control power for the diesel generators in the emergency AC electrical system. Figure 3.15 is a simplified schematic of the DC system.

The DC system is composed of two separate trains, each comprising a 125-volt battery, bus, and control panels. In addition to the batteries, the DC system is also supplied from the emergency AC electrical system via three battery chargers, with two in service at any given time. The busses cannot be cross-tied.

Reactor Building Spray Injection/Reactor Building Spray Recirculation

The reactor building spray system (RBSS) reduces post-accident pressure to nearly atmospheric pressure, removes heat from the containment during recirculation, and removes radioactivity from the containment atmosphere. Figures 3.16 and 3.17 are simplified schematics of the RBSS.

The RBSS consists of two independent trains. Suction sources are the borated water storage tank during injection and the reactor building sump during recirculation. Each train contains a pump with its associated piping and valves.



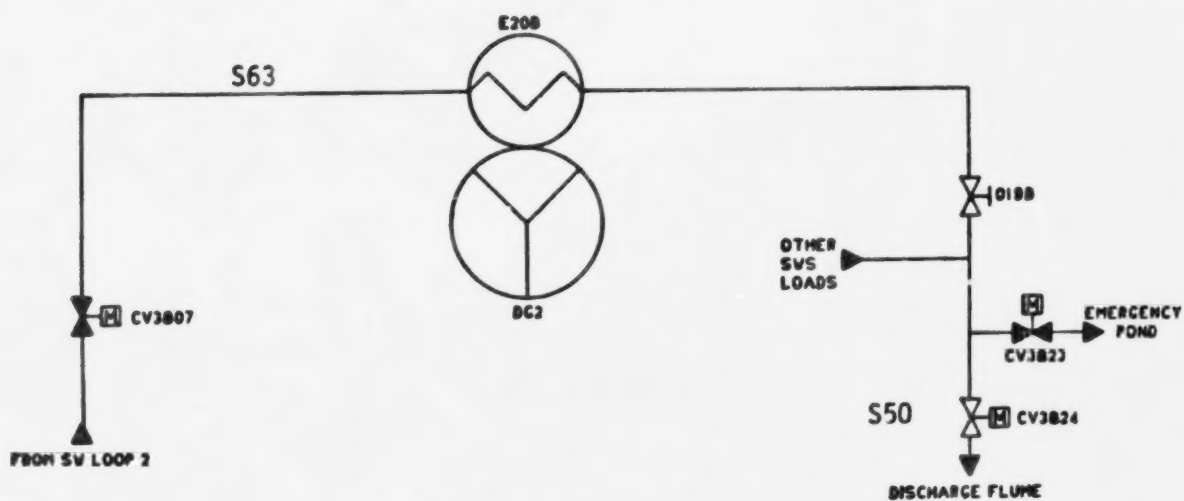
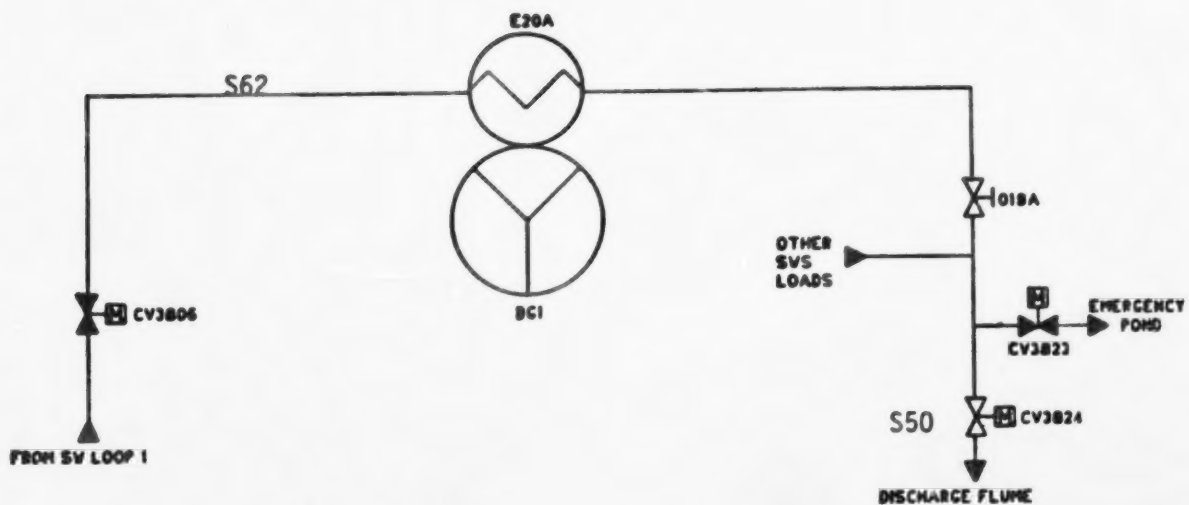


Figure 3-14 Pipe Segments for Service Water Cooling of Diesel Generators

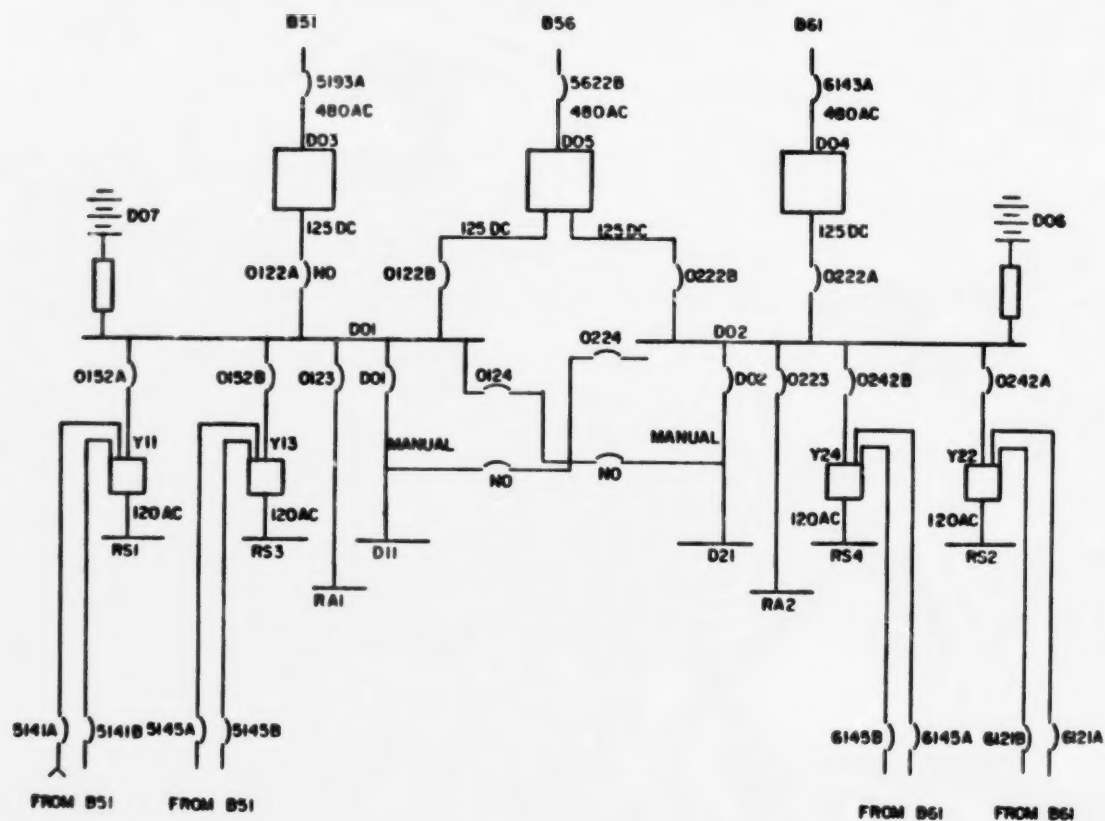


Figure 3-15 DC Power System

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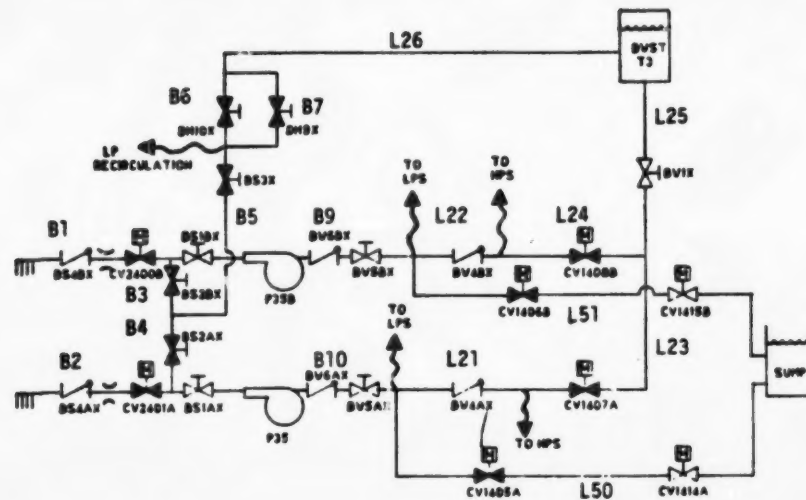


Figure 3-16 Pipe Segments for the Reactor Building Spray Injection/Recirculation Systems

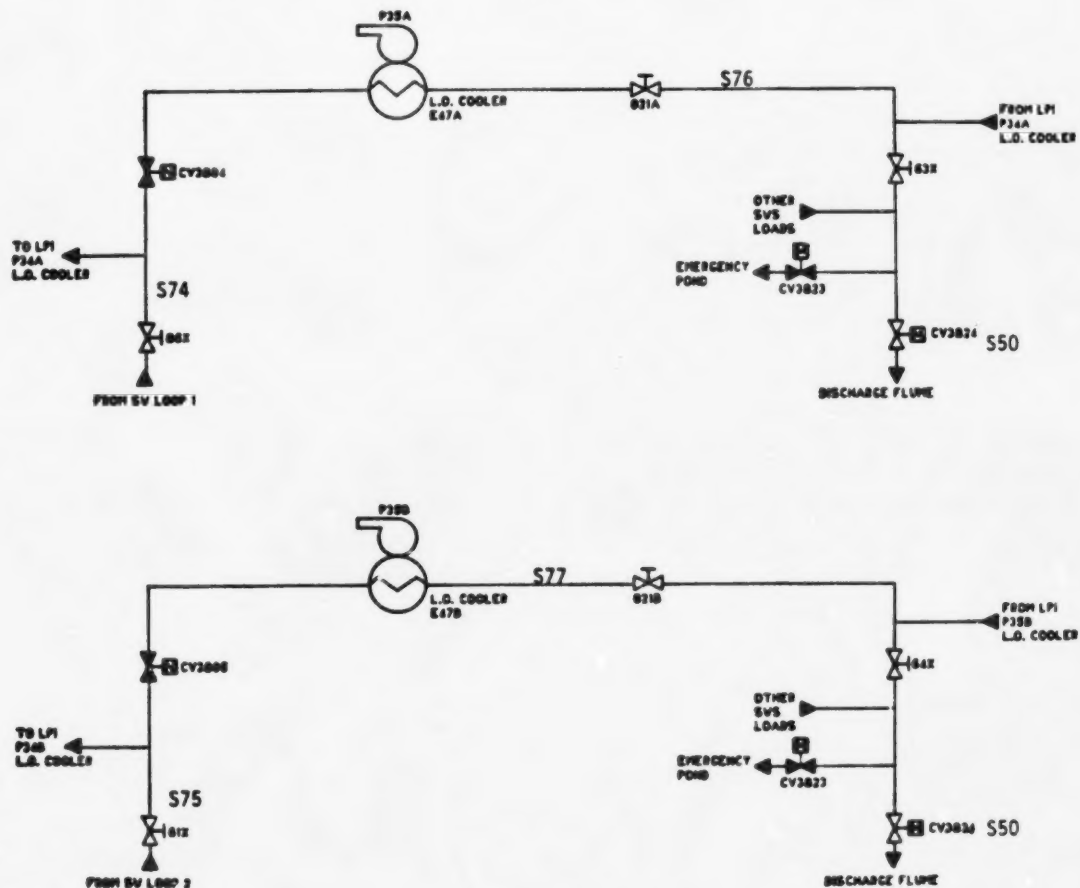


Figure 3-17 Pipe Segments for Reactor Building Spray Pumps Lube Oil Cooling

4.0 SEISMICITY AND HAZARD CURVES

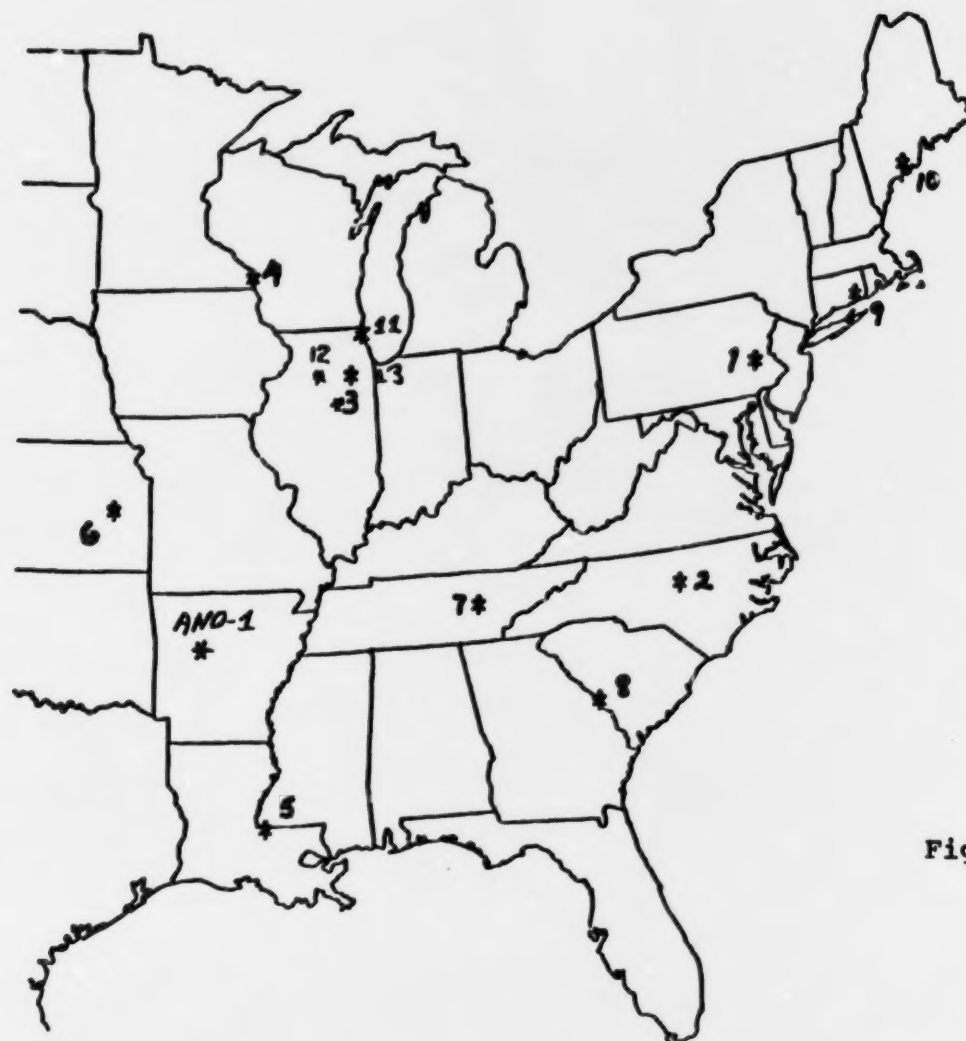
The ANO-1 Nuclear Power Plant is located in northwestern Arkansas and is founded directly on shale rock. The location of the plant is generally considered to be in the Central Stable Zone of seismicity. The proximity to the New Madrid site results in a relatively high SSE equal to 0.20g.

Nearby sites for which hazard curve determinations have been made to include Wolf Creek, Braidwood, Watts Bar, and River Bend, as shown on Figure 4.1. River Bend is the most similar site, and since a hazard curve has been developed for the River Bend location,⁹ the hazard curve for ANO-1 is based on the River Bend hazard curve. The ANO-1 hazard curve was obtained by shifting the River Bend (rock outcrop) hazard curve vertically upward so that the 0.20g SSE at ANO-1 is fixed at a probability of exceedance of $2.5E-4$ per year. The River Bend (rock) hazard curve and the estimated ANO-1 hazard curve are shown on Figure 4.2. For comparison, the hazard curves for the next nearest plants (Braidwood, Wolf Creek, and Watts Bar) are also plotted on Figure 4.2. (Note that, due to the similarity of slopes of the four hazard curves shown, it would have made little difference which curve was shifted.) From this figure, probabilities of peak ground accelerations were determined as listed in Table 4.1.

Table 4.1

ANO-1 Hazard Curve Coordinates

<u>ANO-1 PGA</u>	<u>Annual Probability of Exceedance</u>
0.10g	$1.2E-3$
0.20g	$3.0E-4$
0.30g	$1.1E-4$
0.40g	$5.0E-5$
0.50g	$2.8E-5$
0.60g	$1.6E-5$
0.70g	$1.0E-5$
0.80g	$7.3E-6$



Key to Site Index Numbers

1. Limerick
2. Shearon Harris
3. Braidwood
4. La Crosse
5. River Bend
6. Wolf Creek
7. Watts Bar
8. Vogtle
9. Millstone
10. Maine Yankee
11. Zion
12. La Salle
13. Dresden

Figure 4.1 Location of the Sample Sites in the EUS

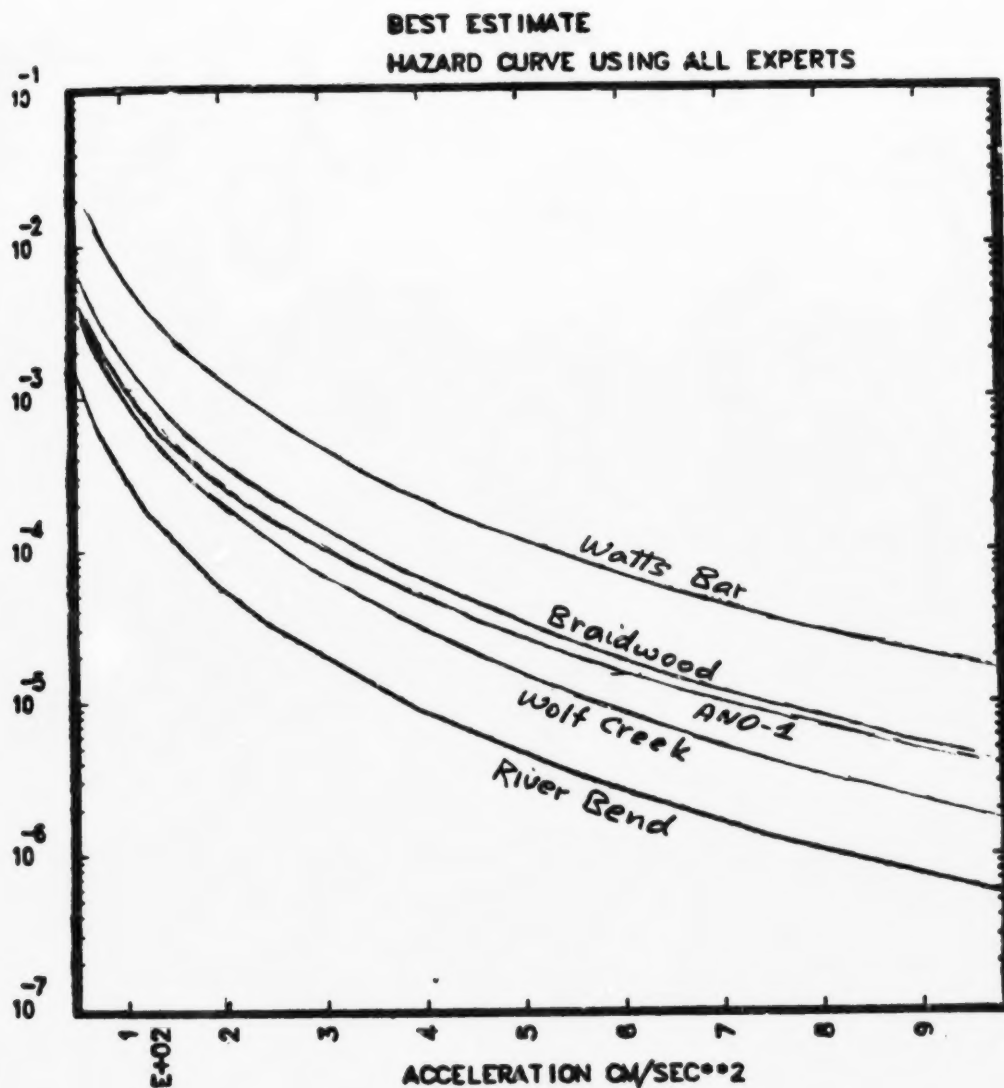


Figure 4.2 Estimated hazard curve for ANO-1 and nearby site hazard curves from Ref. 3.

5.0 RESPONSE CALCULATIONS

5.1 Introduction

As described previously, seismic probabilistic risk assessments (PRA's) can be considered in a series of steps: seismic hazard characterization, seismic response of structures and components, structure and component failure descriptions, plant logic models, and probabilistic failure calculations. Section 5 deals with the frequency characteristics of the free-field ground motion (an element of the seismic hazard characterization) and the seismic response of structures and components.

In a seismic PRA of decay heat removal (DHR) systems, only the components affecting the operation of the systems and those structures housing or supporting these components need to be analyzed. Plant logic models identify the components. Plant general arrangement and mechanical drawings are then used to locate the components and identify the relevant supporting structures. For the Arkansas Nuclear One (ANO) Power Plant, the specific components of concern to the DHR systems and the component locations had not been identified at the time of the seismic analysis. However, previous experience in the analysis of DHR systems at similar plants provided sufficient insight to identify the structures to be analyzed. For this study, seismic analyses of the reactor and auxiliary building were performed. Figure 5.1 illustrates the general plant layout and shows relative location of these structures.

Seismic PRA's require as input best-estimate structural response, variations of response and correlation of response. Recall that the seismic PRA considers earthquakes over the entire range of the seismic hazard curve, hence, seismic responses must be determined over this range. Often, seismic response determined as part of the plant design process is available. However, these data reflect the conservatism associated with the seismic design analysis methodology and consider only low seismic excitation levels.

To determine structural response at the higher excitation levels required by a seismic PRA, either the design analyses must be extrapolated or reanalyses of the structures must be made. For this study, mathematical models of the reactor and auxiliary buildings developed for the design analyses by the utility were used in detailed probabilistic response analysis to determine the best-estimate seismic response of these structures.

The balance of this chapter will describe and summarize

- the site and earthquake characteristics that provide the starting point in the best-estimate soil-structure analysis of the ANO structures
- the probabilistic response analysis of the reactor

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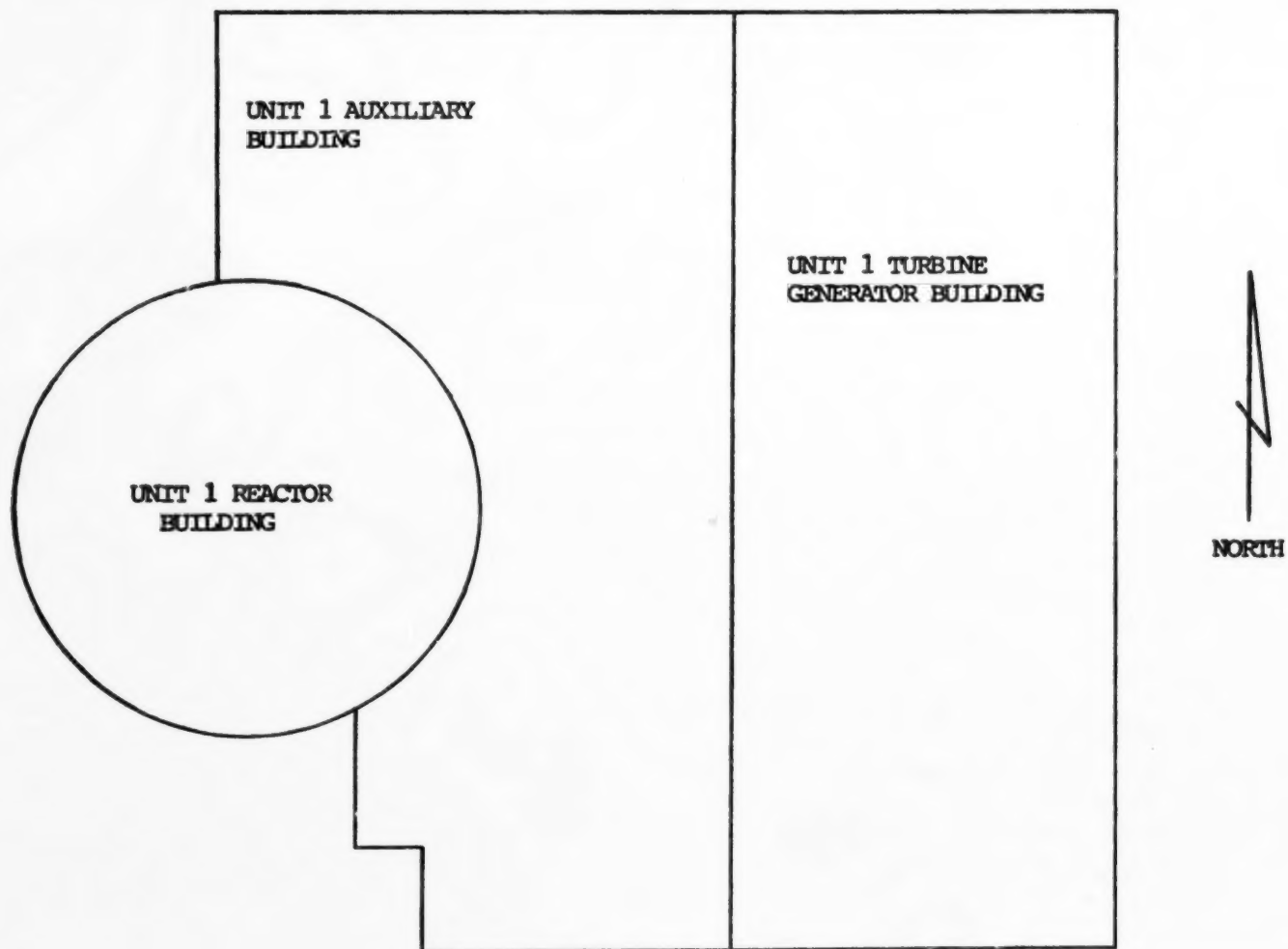


Fig. 5.1 Arkansas Nuclear One General Plant Layout.

building and the auxiliary building

- the in-structure responses that define the response of components associated with the decay heat removal systems

5.2 Site and Earthquake Characteristics

5.2.1 Site Description

The Arkansas Nuclear One site in its natural state is an area where clay and silty clay deposits overlie bedrock consisting of Pennsylvanian McAlester formation shale. Thickness of the clay overburden varies from 13 to 24 feet at the site. Hard fine grained sandstone of the Harthshorne formation underlie the shale and was encountered at a depth of 150 ft. Basemats of the structures of concern to this study were embedded and founded on the Pennsylvanian McAlester shale. Figure 5.2 shows grade and foundation elevations for the ANO reactor and auxiliary buildings.

5.2.2 Soil Properties and Earthquake Definition

Two inter-related objectives of the initial portion of this investigation were to:

- define soil properties as a function of excitation level over the range of earthquakes to be considered, i.e. as defined by the seismic hazard curve.
- define the input motion for the determination of seismic responses of structures and components for this range of earthquakes.

Three earthquake levels were considered and defined by their peak ground acceleration in the horizontal direction -- 0.20g (SSE), 0.40g (2SSE), and 0.60g (3SSE). They are denoted acceleration ranges 1, 2 and 3 in subsequent discussions. These earthquake levels were treated explicitly -- input motions and seismic response for other levels defined by the hazard curve can then be interpolated from the results.

In general, soil properties such as shear modulus and damping are a function of soil strain and consequently a function of earthquake acceleration. With higher excitation levels, soil shear modulus tends to decrease while soil damping tends to increase. The ANO soil properties are characteristic of a rock site. For such stiff soils, the degradation in shear modulus is small and there is little local site amplification of the earthquake ground motion. For this reason, a single nominal soil shear modulus was used in the analyses of the ANO structures for all three excitation levels. However, nominal soil damping did vary from one acceleration level to another. Table 5.1 lists the nominal soil properties used.

The frequency characteristics of the free-field ground motion were defined by a best-estimate median response spectra associated with rock and presented in Ref. 13. For our probabilistic response analysis, a single artificial acceleration time history scaled to the appropriate peak acceleration whose

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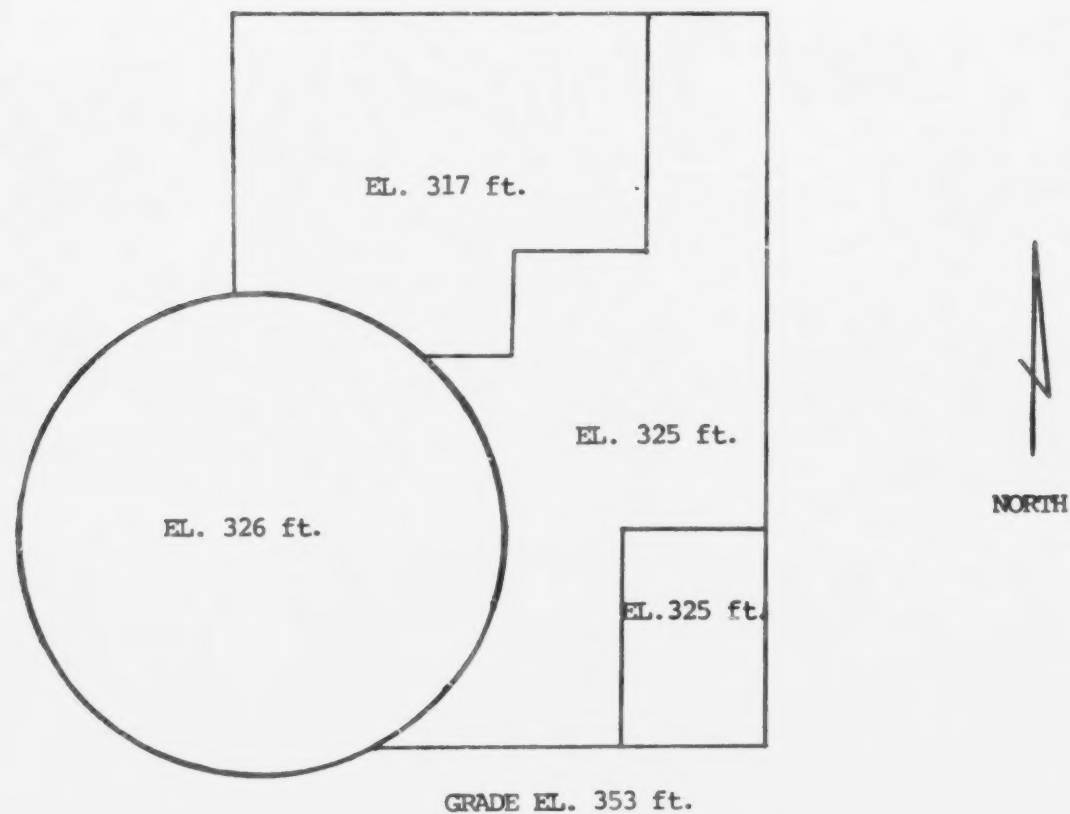


Fig. 5.2 Arkansas Nuclear One Grade and Foundation Elevations for Reactor and Auxiliary Buildings.

Table 5.1 Nominal Soil Properties for ANO-1 Analyses

	<u>Acceleration Range 1</u>	<u>Acceleration Range 2</u>	<u>Acceleration Range 3</u>
Shear Modulus	3.85 E7 lbs/ft ²	3.85 E7 lbs/ft ²	3.85 E7 lbs/ft ²
Damping	7%	8.5%	10%

response spectra approximated this best-estimate spectra defined the free-field motion. Figure 5.3 compares the median rock spectra from Ref. 13 with the response spectra generated from the artificial time history for 5% damping. A single time history (rather than an ensemble of motions) was selected to model the free-field motion for both technical and budgetary reasons. A more thorough approach is to define an ensemble of motions at the free-field to describe the frequency characteristics of the motion -- their median response spectra closely approximating the median of Ref. 13. In the present case, however, a single time history is adequate for our purposes. First, the SSI analyses, to be discussed later, includes variability in soil properties which does introduce variability in the input motion to the structures. Second, the objective of this effort is to predict median seismic response not the entire response distribution, hence, a smaller number of samples is required and a complete specification of variability in the free-field motion is not necessary.

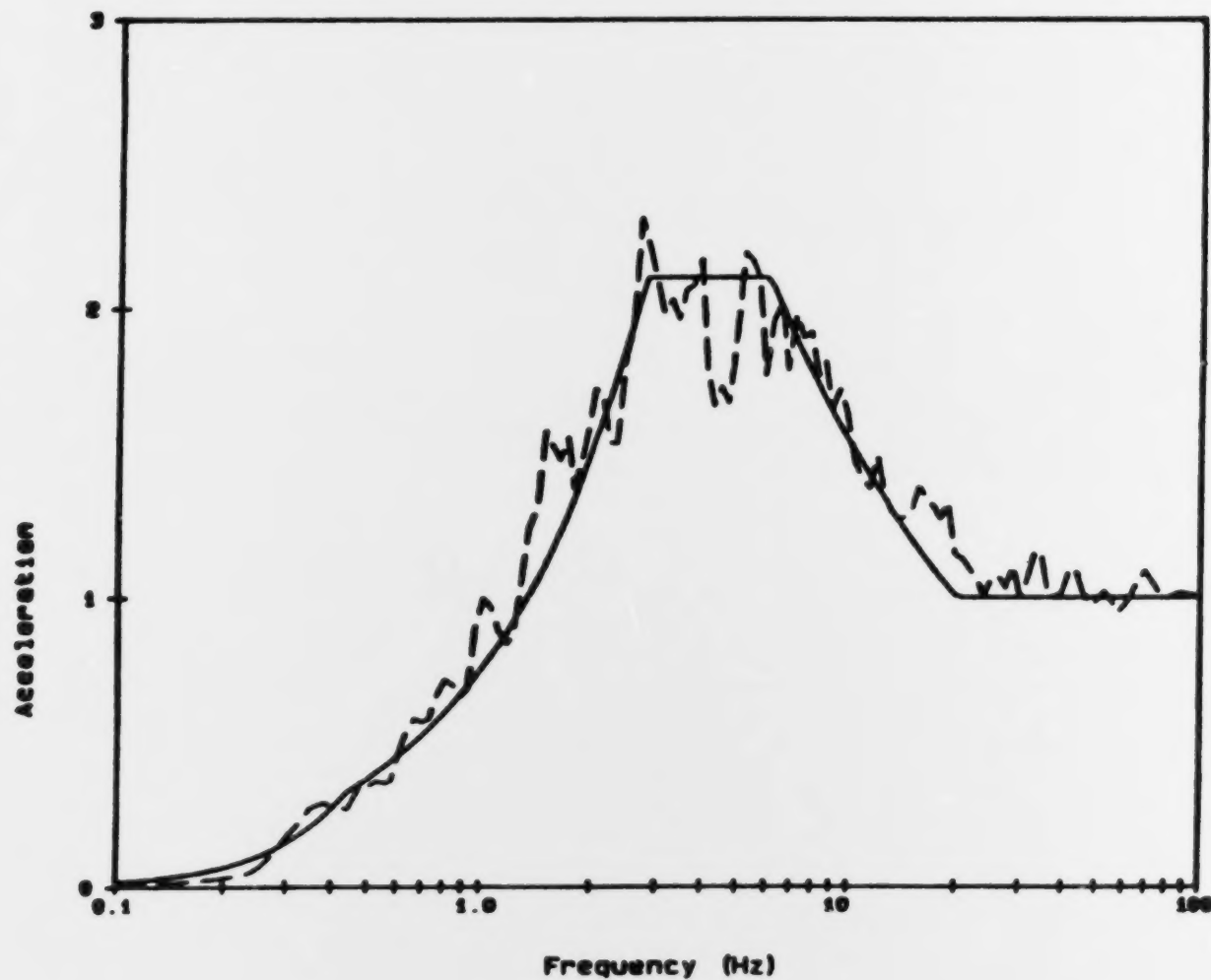


Fig. 5.3 Comparison of Ground Response Spectra on the Rock Outcrop - Median from Ref. 1 vs. Artificial Acceleration Time History - 5% Damping

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5.3 Probabilistic Response Analysis of the Reactor Building and Auxiliary Building

In recognition of the importance of the effects* of embedment and soil structure interaction, probabilistic response analyses to generate mean response for the seismic PRA were performed on the reactor and auxiliary buildings. The methodology used is that of SMACS [15] as implemented in the computer program CLASSI [16] utilizing the substructure approach. The substructure approach to SSI is composed of the following elements: specification of the free-field ground motion; calculation of the dynamic characteristics of the structures; determination of the foundation impedances and analysis of the coupled soil structure system.

The procedure is to perform a series of deterministic analyses, each simulating an earthquake occurrence, including variability in seismic input, soil-structure interaction, and in-structure representation.

The seismic input variability is normally introduced by considering an ensemble of earthquake motions. For this study, only the one artificial earthquake motion described earlier was used. Note that a series of ten earthquake simulations for each acceleration range were performed but that each used the identical free-field input motion as a starting point.

Soil structure interaction and structure response variability are introduced through a limited number of parameters -- soil shear modulus, soil damping, structure frequency and modal damping. Variability in SSI was incorporated through modeling soil shear modulus and material damping as random variables with lognormal distributions -- medians corresponding to the nominal values of Table 5.1 and coefficients of variation of 0.4 and 0.5, respectively. Variability in structure dynamic behavior was also modeled by treating structure frequency and modal damping as random variables. Nominal values of structure frequencies were those calculated in the eigenvalue extraction analysis for the structure model. Nominal values of structure damping were taken to be 0.07, 0.085, and 0.10 (fractions of critical damping) for the three seismic acceleration ranges considered here. These were based on published damping values and assumed stress levels achieved.

Parameter variations in each step of the response analysis were selected to represent random variability, not to include modeling uncertainty. The assumed parameter variability corresponds to that of the SSMRP [3]. The parameter values for each of the ten simulations were selected from the probability distributions by dividing the distributions into equally probable segments, sampling from each segment, and combining the samples by a Latin hypercube experimental design.

The responses calculated from the simulations are combined

to estimate mean responses conditional on the occurrence of an earthquake described by a particular hazard curve parameter, e.g. peak ground acceleration.

Reactor Building Internal Structure Response Analyses The reactor building is a reinforced concrete building, circular in plan (~ 61.75 ft. radius). The building is supported on a 8.5 ft. thick reinforced concrete mat at elevation 326 ft. and extends in height to elevation 535 ft. Grade is at 353 ft.

The dynamic model of the reactor building used in the response analyses is a 2-D lumped mass model documented in Ref. 8. A fixed-base model excluding the soil springs was reconstructed from the given model data. Figure 5.4 shows the model and correlates node points and floor elevations. The internal structure and containment shell are modeled without any coupling between them. Response points associated with DHR components of concern are located within the internal structure. The internal structures fundamental mode of vibration has a frequency of 9.91 Hz. and accounts for 85% of the mass participating in the horizontal direction.

For each of the ten earthquake simulations at each of the internal structures nodes, acceleration time histories and corresponding in-structure response spectra at 5% damping were calculated. These spectra were then combined into mean responses for all nodes in the internal structure model (nodes 6-9). Figure 5.5 presents mean horizontal in-structure spectra for the three seismic acceleration levels. Each graph shows an overplot of the mean spectrum at the reactor building foundation (node 6) and the mean spectra for the internal structure (nodes 7-9). These spectra formed the basis for determining the response of decay heat removal components housed at the various elevations in the internal structure.

Auxiliary Building Response Analyses The auxiliary building is a reinforced concrete building, rectangular in plan (143 ft. x 175 ft.). The building is supported on mat foundations at elevations 335 ft. and 317 ft. An equivalent depth of embedment based on surface area at the two depths is 27 ft. The amount of embedment is significant for the SSI analyses.

The dynamic model of the auxiliary building is a 2-D lumped mass stick model documented in Ref. 8. As for the reactor building, a fixed-base model of the auxiliary building excluding the soil springs was reconstructed from the given data and used for the probabilistic response analyses. Figure 5.6 shows the model and correlates node points and floor elevations. The auxiliary building fundamental mode of vibration has a frequency of 10.74 Hz. This mode accounts for 91.5% of mass participating in the horizontal direction.

For each of the earthquake simulations, acceleration time histories and corresponding in-structure response spectra at 5%

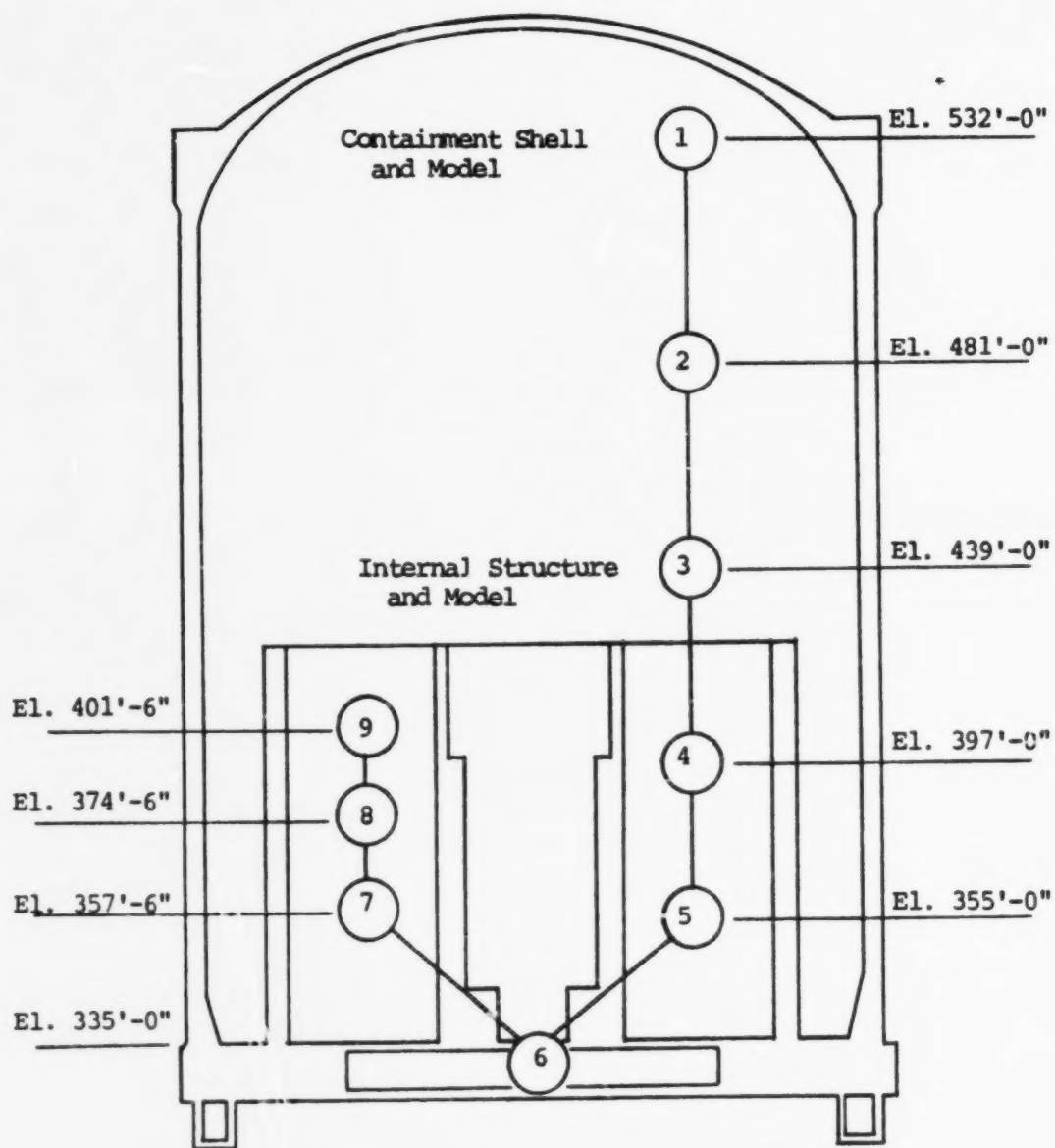


Fig. 5.4 Arkansas Nuclear One Reactor Building Structural Model

ano-1: mean of 10 node 6 .20g
 ano-1: mean of 10 node 7 .20g
 ano-1: mean of 10 node 8 .20g
 ano-1: mean of 10 node 9 .20g

5.00 pct damping —————
 5.00 pct damping - - - - -
 5.00 pct damping
 5.00 pct damping — · — · —

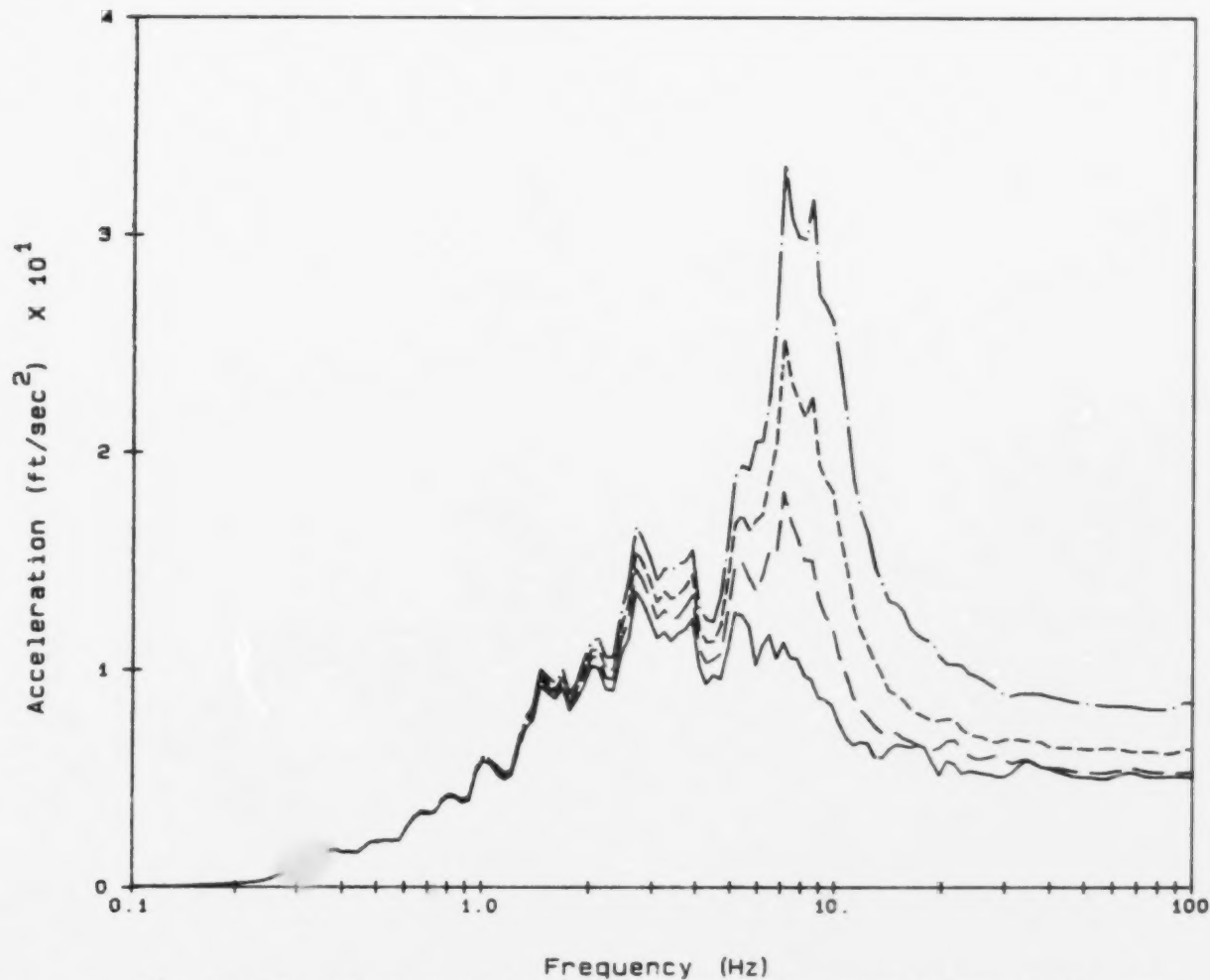


Fig. 5.5 a) Arkansas Nuclear One Internal Structure Response, .20g Free-field ZPA.

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ano-1; mean of 10 node 6 .40g
 ano-1; mean of 10 node 7 .40g
 ano-1; mean of 10 node 8 .40g
 ano-1; mean of 10 node 9 .40g

5.00 pct damping —————
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 5.00 pct damping - - - - -
 5.00 pct damping —————

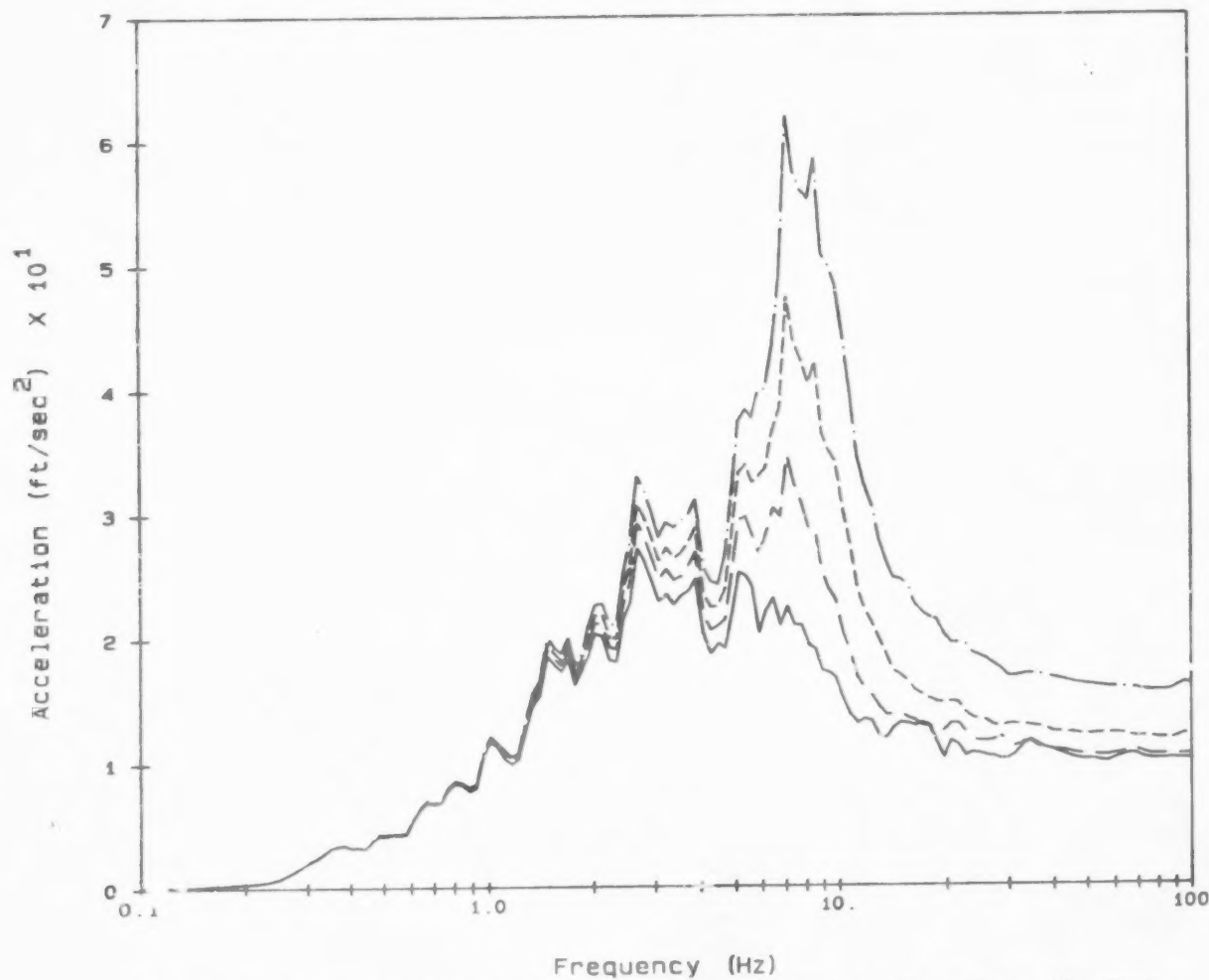


Fig. 5.5 b) Arkansas Nuclear One Internal Structure Response, .40g Free-field ZPA.

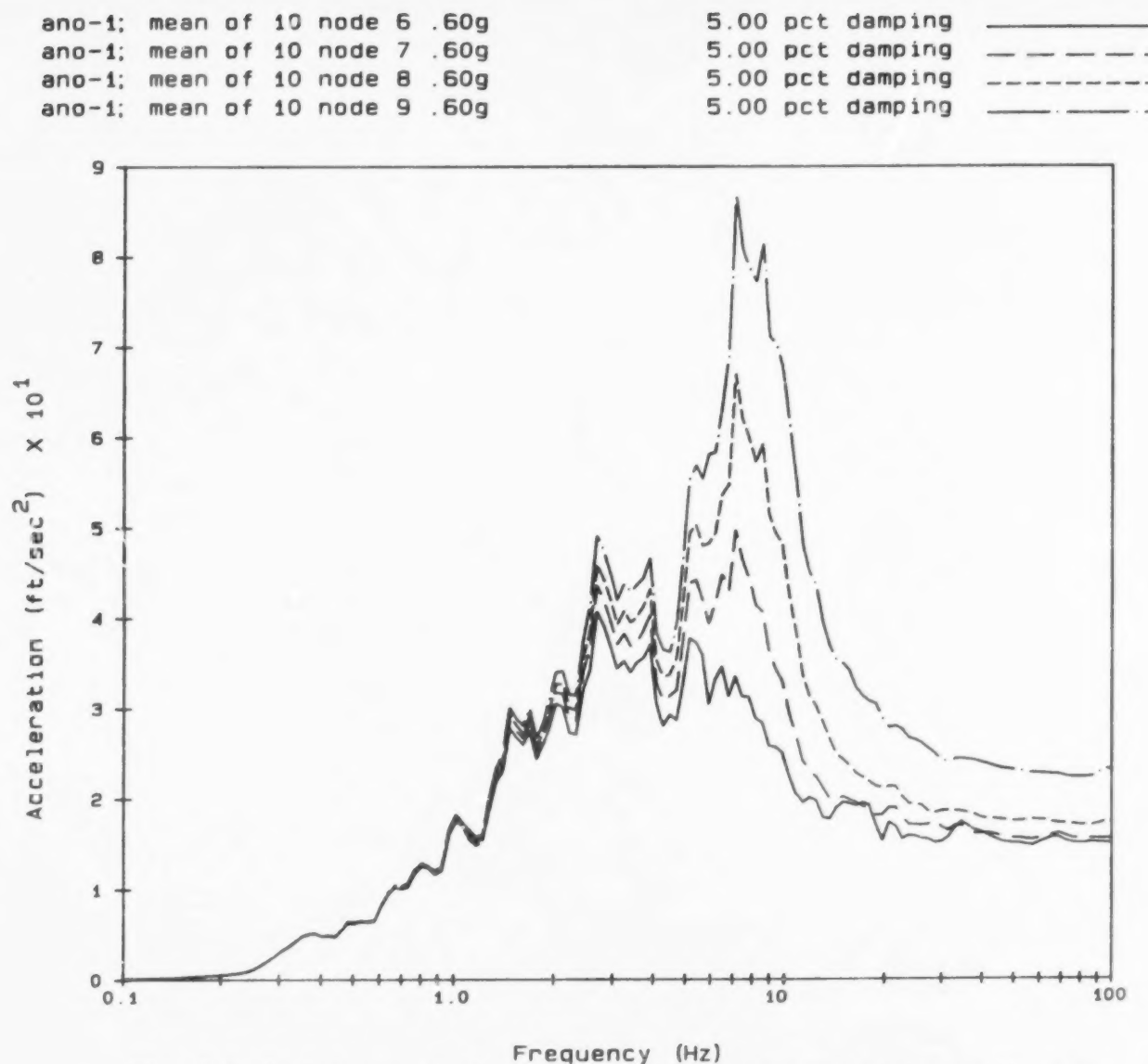


Fig. 5.5 c) Arkansas Nuclear One Internal Structure Response, .60g Free-field ZPA.

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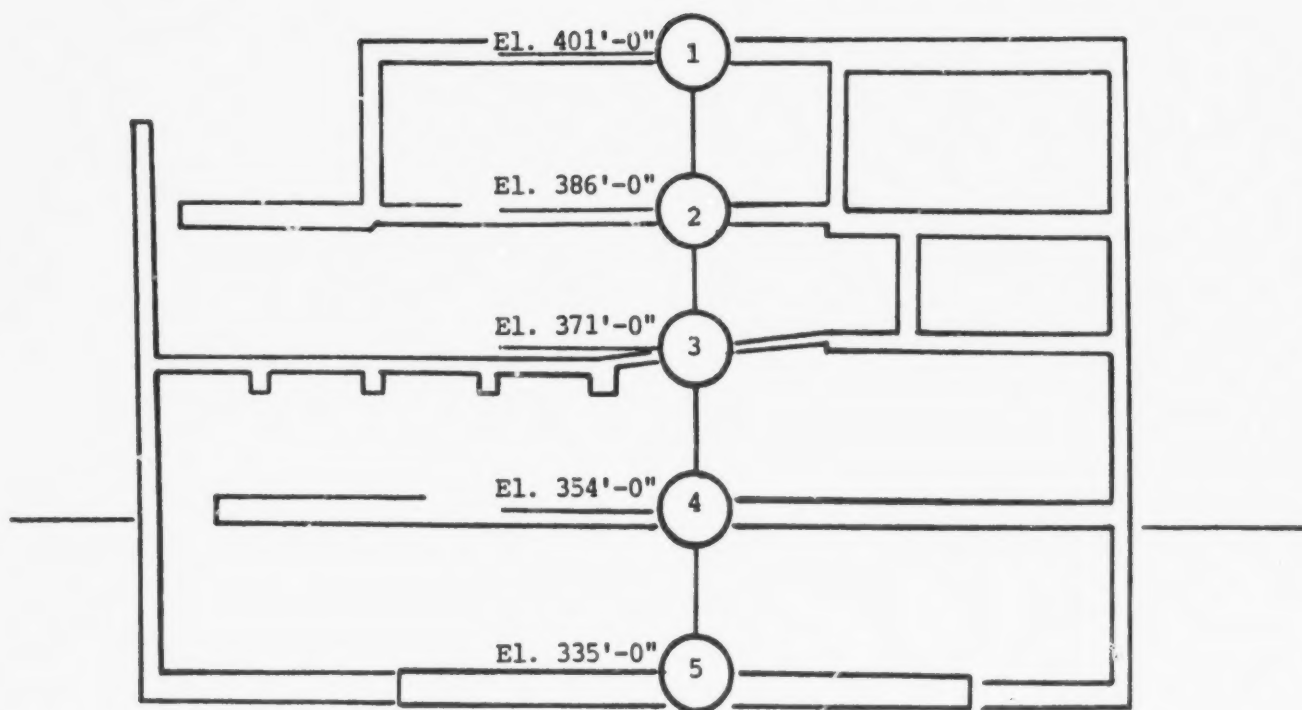


Fig. 5.6 Arkansas Nuclear One Auxiliary Building Structural Model.

damping were calculated. These spectra were then combined into mean responses for all nodes. Figure 5.7 presents mean horizontal spectra for the three seismic acceleration levels. Each plot shows an overplot of the mean in-structure spectrum at the reactor building foundation (node 5) and the mean in-structure spectra for the auxiliary building (nodes 1-4). These spectra formed the basis for determining the response of decay heat removal components housed in the auxiliary building.

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ano-1: mean of 10 node 1 .20g
 ano-1: mean of 10 node 2 .20g
 ano-1: mean of 10 node 3 .20g
 ano-1: mean of 10 node 4 .20g
 ano-1: mean of 10 node 5 .20g

5.00 pct damping —————
 5.00 pct damping - - - - -
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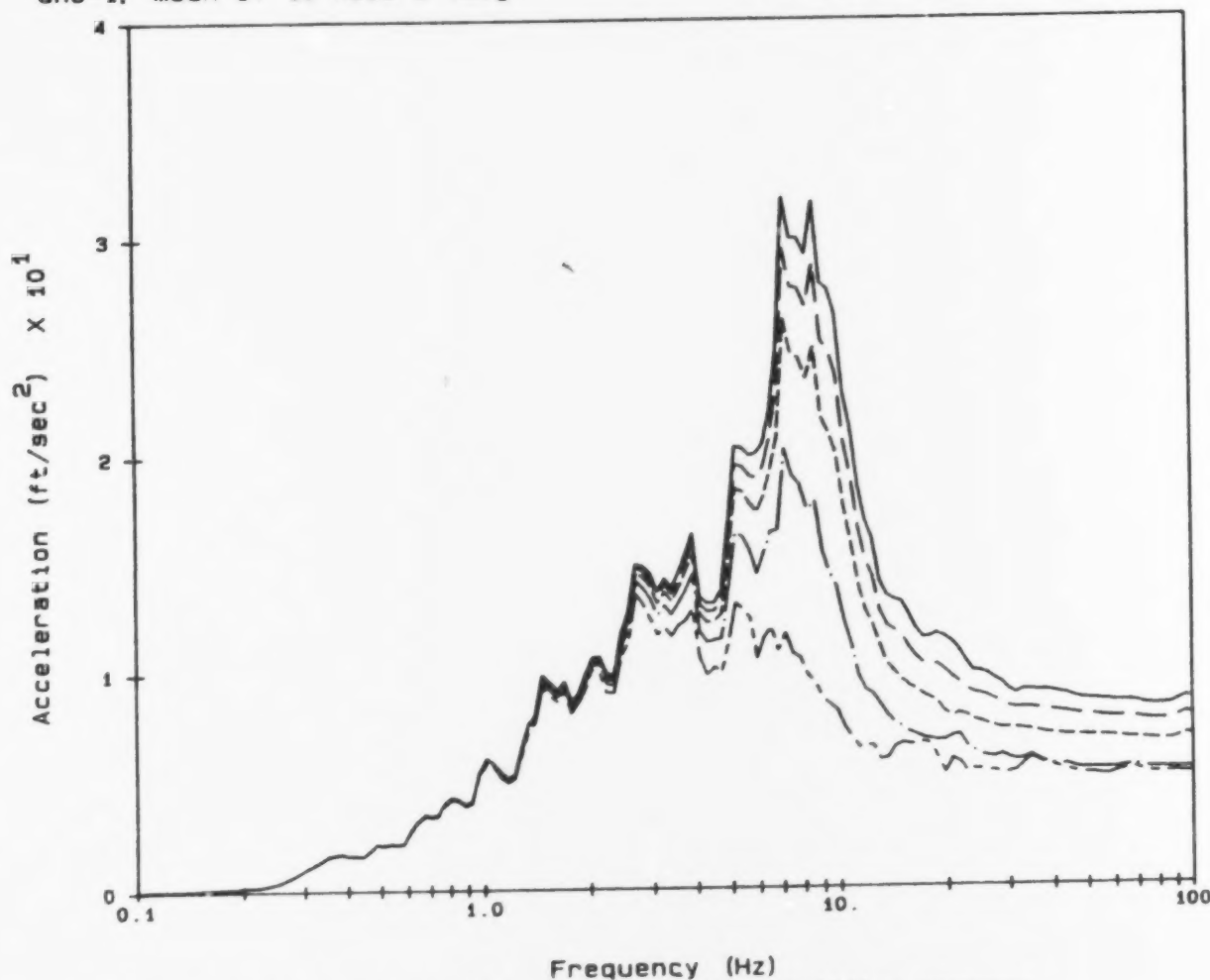


Fig. 5.7 a) Arkansas Nuclear One Auxiliary Building Response,
 Acceleration Range 1.

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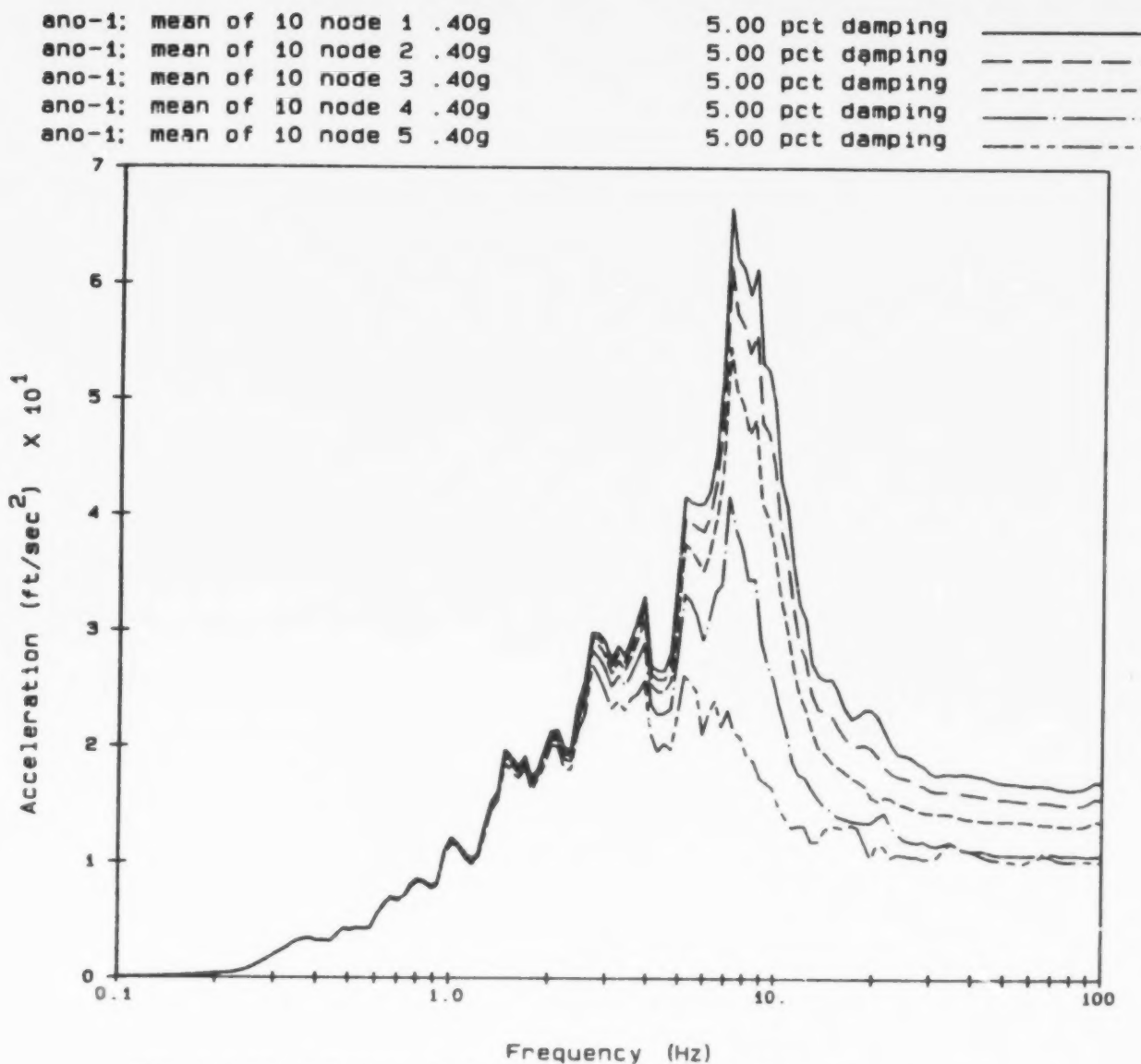


Fig. 5.7 b) Arkansas Nuclear One Auxiliary Building Response,
Acceleration Range 2.

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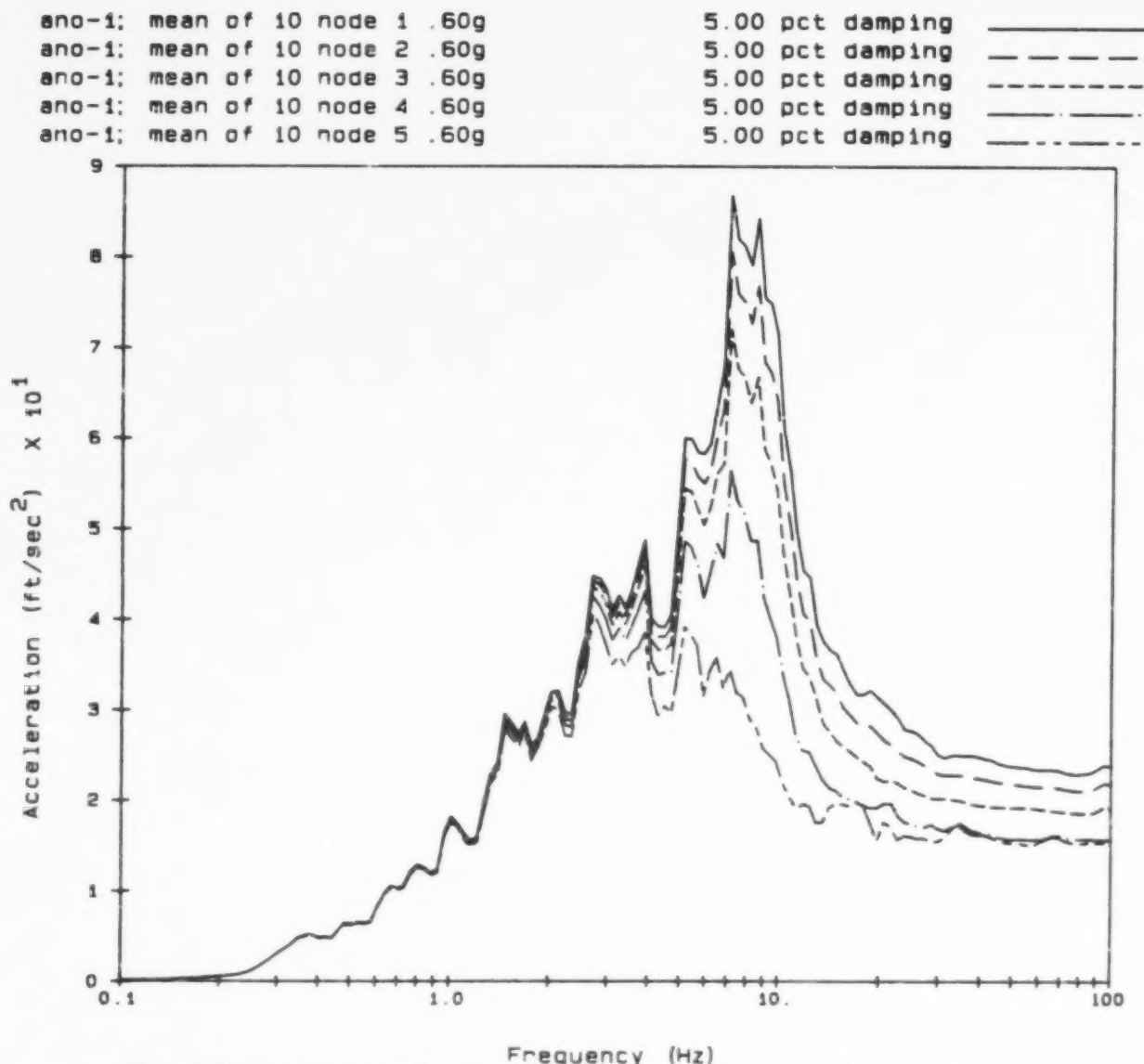


Fig. 5.7 c) Arkansas Nuclear One Auxiliary Building Response,
Acceleration Range 3.

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5.4 Decay Heat Removal (DHR) System Component Responses

The in-structure spectra presented and discussed in the previous section are used to determine DHR component response. Assuming that the dynamic characteristics of a given component can be represented by a single dominant mode of vibration, the component response can be approximated by the spectral acceleration of the appropriate in-structure spectra at the frequency of the dominant mode.

For the three levels of ground motion considered in this study, best estimate DHR systems component responses are presented in Table 5.2. At each structural location, response is calculated over several frequency ranges. These ranges span the probable natural frequencies of the components housed at that location. The mean zero period acceleration response is calculated from the ten values given by the probabilistic response analysis. The mean response over a frequency range is calculated by determining the spectral accelerations at 25 points over the range from the mean spectra given by the ten earthquake simulations. These spectral accelerations in turn are used to determine the mean response. Given the natural frequency of the component of interest, the appropriate frequency interval and component response is then defined.

Table 5.2 ANO-1 Seismic Responses

<u>Location</u>	<u>Elevation</u>	<u>Frequency</u>	Response for Given Peak Ground Acceleration (G's)		
			<u>0.20</u>	<u>0.40</u>	<u>0.60</u>
RB	335.	2-5	.34	.68	1.02
RB	335.	5-10	.32	.64	.95
RB	335.	10-20	.20	.40	.59
RB	335.	ZPA	.16	.31	.47
RB	357.5	2-5	.37	.74	1.11
RB	357.5	5-10	.45	.88	1.27
RB	357.5	10-20	.23	.45	.66
RB	357.5	ZPA	.16	.33	.49
RB	374.5	2-5	.40	.79	1.18
RB	374.5	5-10	.61	1.16	1.67
RB	374.5	10-20	.31	.59	.84
RB	374.5	ZPA	.20	.38	.54
RB	401.5	2-5	.43	.86	1.28
RB	401.5	5-10	.80	1.52	2.15
RB	401.5	10-20	.45	.84	1.19
RB	401.5	ZPA	.26	.50	.72
AB	401.	2-5	.43	.86	1.28
AB	401.	5-10	.80	1.61	2.22
AB	401.	10-20	.45	.87	1.22
AB	401.	ZPA	.26	.53	.74

Table 5.2 ANO-1 Seismic Responses (cont.)

<u>Location</u>	<u>Elevation</u>	<u>Frequency</u>	Response for Given Peak Ground Acceleration (G's)		
			<u>0.20</u>	<u>0.40</u>	<u>0.60</u>
AB	386.	2-5	.42	.84	1.25
AB	386.	5-10	.74	1.48	2.06
AB	386.	10-20	.40	.77	1.09
AB	386.	ZPA	.24	.40	.68
AB	371.	2-5	.41	.82	1.21
AB	371.	5-10	.66	1.31	1.83
AB	371.	10-20	.33	.65	.93
AB	371.	ZPA	.21	.42	.60
AB	354.	2-5	.38	.77	1.15
AB	354.	5-10	.51	1.01	1.44
AB	354.	10-20	.25	.49	.71
AB	354.	ZPA	.17	.33	.49
AB	335.	2-5	.35	.70	1.04
AB	335.	5-10	.32	.65	.96
AB	335.	10-20	.20	.40	.59
AB	335.	ZPA	.16	.32	.47
Yard	353.	2-5	.38	.76	1.14
Yard	353.	5-10	.37	.75	1.12
Yard	353.	10-20	.27	.53	.80
Yard	353.	ZPA	.20	.40	.60

AB = Auxiliary Building
RB = Reactor Building

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5.5 Responses in Terms of Peak Ground Acceleration

The responses in Table 5.2 are given at three rock outcrop accelerations (0.20g, 0.40g, and 0.60g). Figures 5.8 through 5.10 are plots of the response point accelerations versus the peak ground acceleration level. It can be seen that a linear relation exists up to free-field accelerations of 0.60g or greater. Furthermore, for those curves which show significant non-linearity at higher acceleration levels, the linear relation provides a conservative estimate of the local response. Such behavior was first recognized in the SSMRP studies.³ Inasmuch as the bulk of the risk is generally due to earthquake levels only up to the 4 SSE level, it can be seen that relating local response to peak ground acceleration in a linear fashion is appropriate for the ANO-1 analysis. From these figures, ratios between the various responses and the surface peak ground acceleration were determined, as listed on Table 5.3. Table 5.4 lists all the basic event component failures, their locations, response numbers and fragility categories. These two tables can be used in conjunction with tables of the generic component fragility values (Attachment C.1) to obtain the component failure probabilities directly. Attachment C.3 lists base case plant specific failure probabilities which are the sum of random and seismic failures.

5.6 Variability in Response

Variability in responses (floor slab and spectral accelerations) is assigned based on the detailed SSMRP response calculations made for the Zion Nuclear Power Plant.³ In this evaluation, an attempt was made to accurately compute the responses of walls and floor slabs in the Zion structures, moments in the important piping systems, accelerations of all important valves, and the spectral accelerations at each safety system component (pump, electrical bus, motor control center, etc.). Correlation between the responses of all components was computed directly from the dynamic response calculations. Detailed summary plots showed that the uncertainty in the various responses could be reasonably approximated as follows:

<u>Response Type</u>	<u>β random</u>
Peak Ground Acceleration	0.25
Floor Zero Period Acceleration	0.35
Floor Spectral Acceleration	0.45

where β is the standard deviation of the logarithm of the response. These values were used in computation of all the generic component failure probabilities in Attachment C.1.

5.7 Correlations Between Responses

Generic rules for estimating correlations between the various

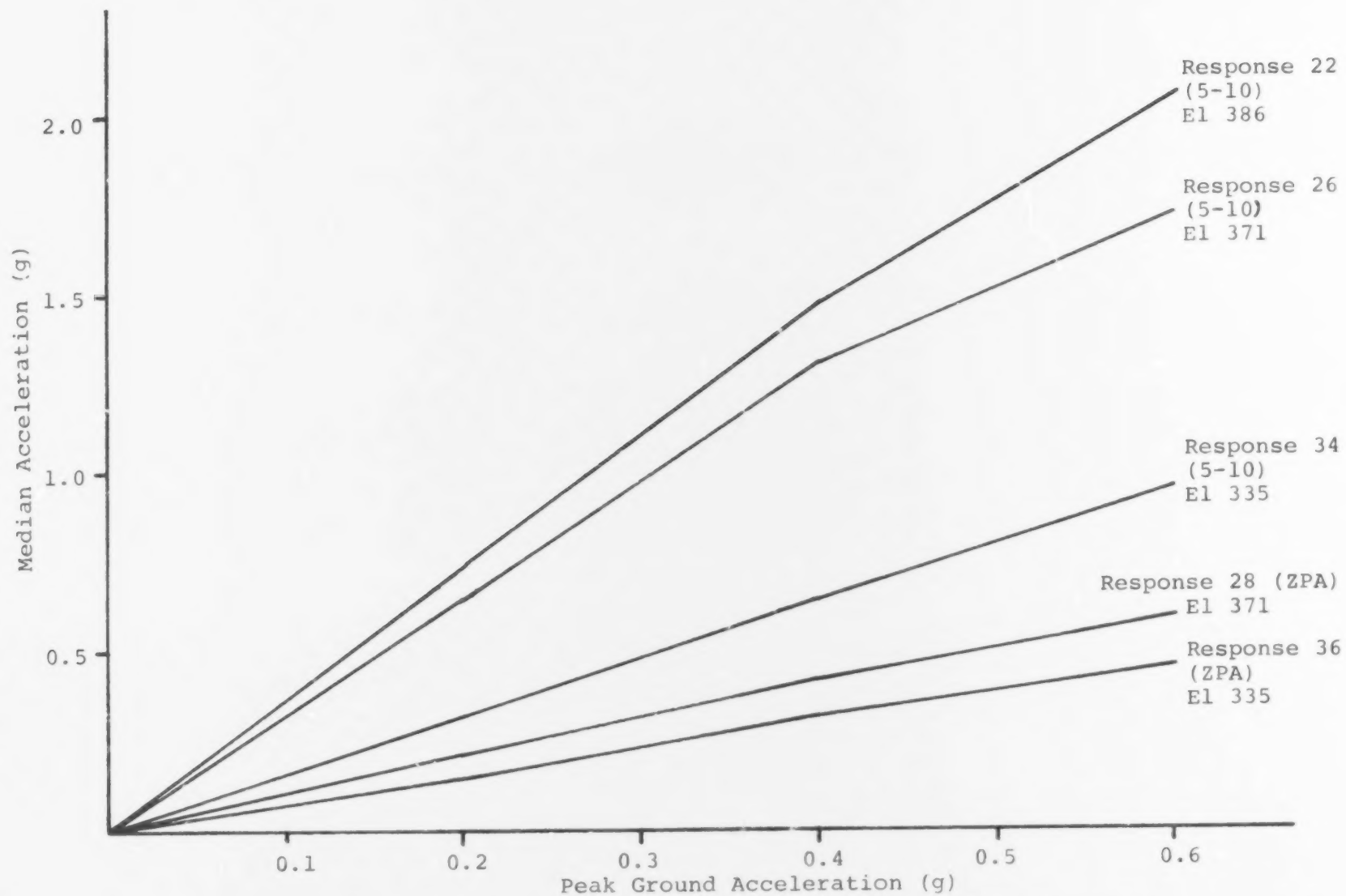


Figure 5.8 Arkansas Nuclear One Auxiliary Building Responses

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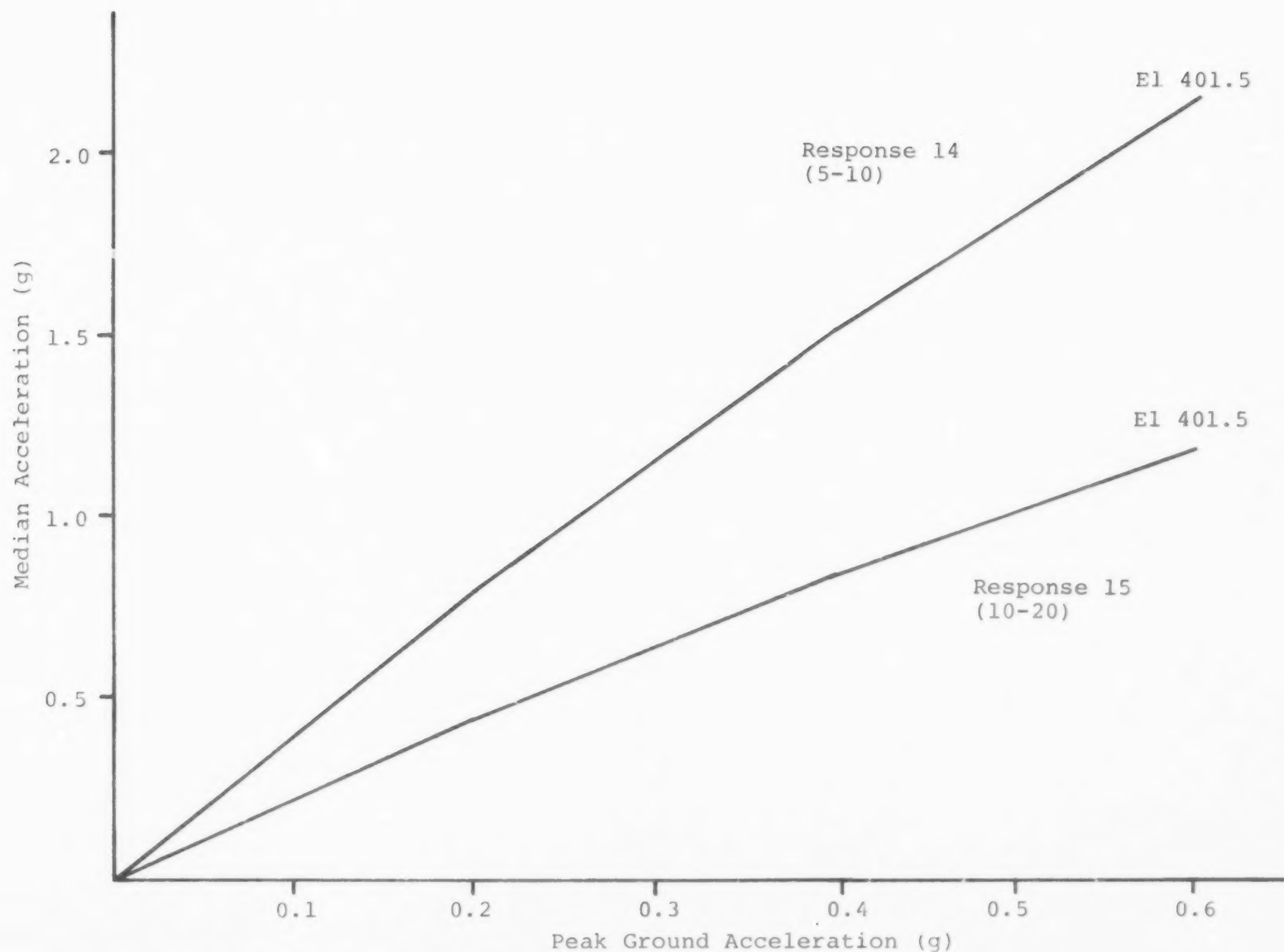


Figure 5.9 Arkansas Nuclear One Reactor Building Responses

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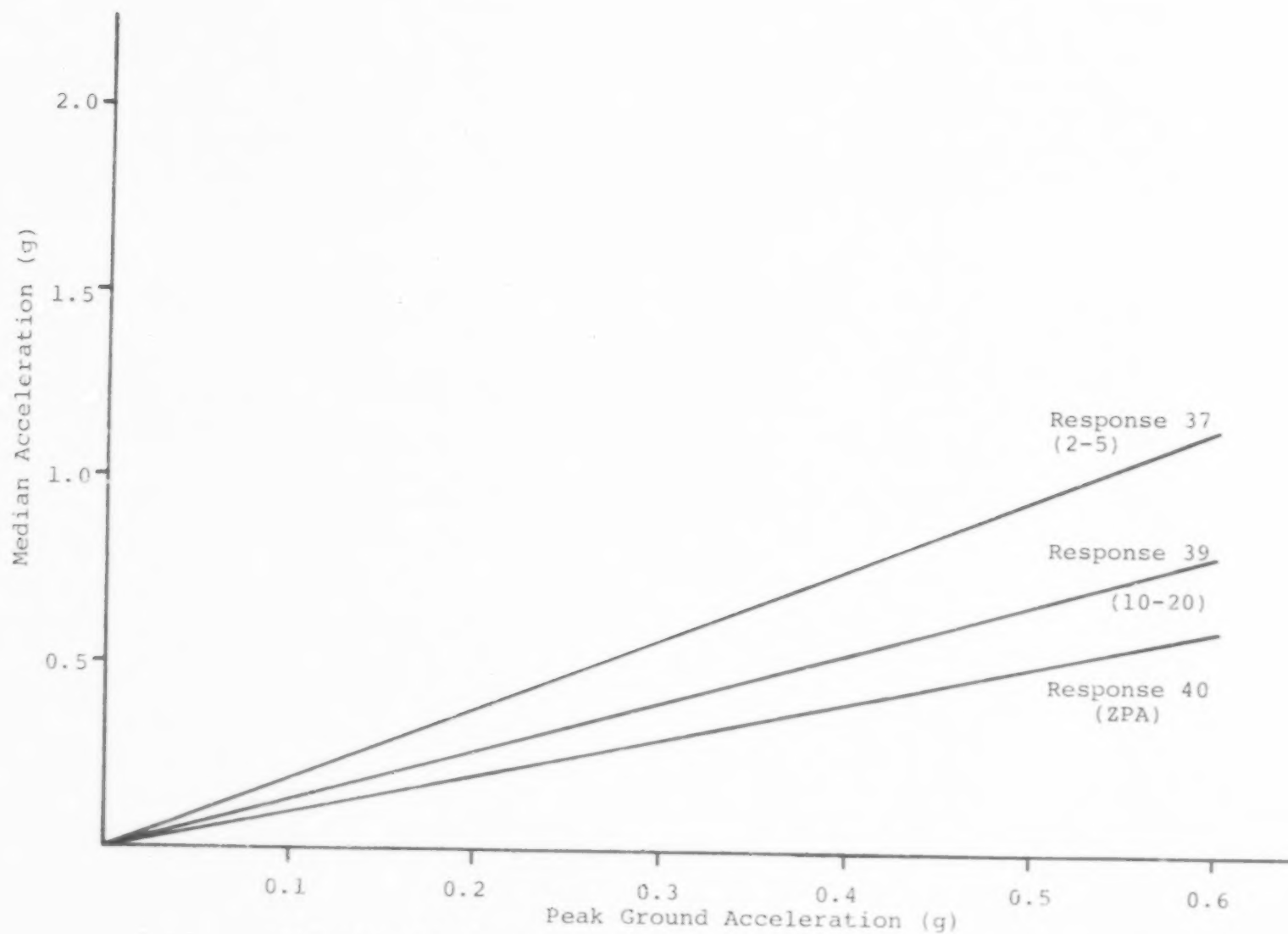


Figure 5.10 Arkansas Nuclear One Yard (Elev 353) Responses

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Table 5.3 ANO-1 Seismic Response Locations

<u>Response Number</u>	<u>Location</u>	<u>Elevation</u>	<u>Frequency</u>	<u>Multiple of PGA</u>
1	RB	335.	2-5	*
2	RB	335.	5-10	*
3	RB	335.	10-20	*
4	RB	335.	ZPA	*
5	RB	357.5	2-5	*
6	RB	357.5	5-10	*
7	RB	357.5	10-20	*
8	RB	357.5	ZPA	*
9	RB	374.5	2-5	*
10	RB	374.5	5-10	*
11	RB	374.5	10-20	*
12	RB	374.5	ZPA	*
13	RB	401.5	2-5	*
14	RB	401.5	5-10	3.75
15	RB	401.5	10-20	2.00
16	RB	401.5	ZPA	*
17	AB	401.	2-5	*
18	AB	401.	5-10	*
19	AB	401.	10-20	*
20	AB	401.	ZPA	*

*Not used.

Notation: AB Auxiliary Building
RB Reactor Building

Table 5.3 ANO-1 Seismic Response Locations (cont.)

<u>Response Number</u>	<u>Location</u>	<u>Elevation</u>	<u>Frequency</u>	<u>Multiple of PGA</u>
21	AB	386.	2-5	*
22	AB	386.	5-10	1.75
23	AB	386.	10-20	*
24	AB	386.	ZPA	*
25	AB	371.	2-5	*
26	AB	371.	5-10	3.25
27	AB	371.	10-20	*
28	AB	371.	ZPA	1.00
29	AB	354.	2-5	*
30	AB	354.	5-10	*
31	AB	354.	10-20	*
32	AB	354.	ZPA	*
33	AB	335.	2-5	*
34	AB	335.	5-10	1.50
35	AB	335.	10-20	*
36	AB	335.	ZPA	0.75
37	Yard	353.	2-5	2.00
38	Yard	353.	5-10	*
39	Yard	353.	10-20	1.25
40	Yard	353.	ZPA	1.00

*Not used.

Notation: AB Auxiliary Building
RB Reactor Building

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Table 5.4 Arkansas Nuclear One Seismic Failure Basic Event Data

<u>Basic Event</u>	<u>Description</u>	<u>Location/Elev.</u>	<u>Fragility Category</u>	<u>Estimated Frequency</u>	<u>Response Number</u>	<u>Comment</u>
EFWOP7AX-PTD-LF	EFW Turbine Pump	AB/335'	16	7	34	
EFWOP7BA-PMD-LF	EFW Motor Pump	AB/335'	16	7	34	
HPIIP36AA-PMD-LF	HPI Pump A	AB/335'	16	7	34	
HPIIP36BA-PMD-LF	HPI Pump B	AB/335'	16	7	34	
HPIIP36CB-PMD-LF	HPI Pump C	AB/335'	16	7	34	
LPIOP34A-PMD-LF	RHR Pump A	AB/317'	16	7	34	
LPIOP34B-PMD-LF	RHR Pump B	AB/317'	16	7	34	
RBIP35AA-PMD-LF	RBI Pump A	AB/317'	16	7	34	
RBIP35BB-PMD-LF	RBI Pump B	AB/317'	16	7	34	
SWSOP4BA-PMD-LF	SWS Pump B	SW Pump Bldg./353'	17	5	37	
SWSOP4CB-PMD-LF	SWS Pump C	SW Pump Bldg./353'	17	5	37	
A1-4160VAC-LF	4160 VAC	TB/371'	Site Specific	5-10	26	
A2-4160VAC-LF	Buses	TB/371'	Site Specific	5-10	26	
A3-4160VAC-LF		AB/371'	Site Specific	5-10	26	
A4-4160VAC-LF		AB/371'	Site Specific	5-10	26	

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Table 5.4 Arkansas Nuclear One Seismic Failure Basic Event Data (Cont.)

<u>Basic Event</u>	<u>Description</u>	<u>Location/Elev.</u>	<u>Fragility Category</u>	<u>Estimated Frequency</u>	<u>Response Number</u>	<u>Comment</u>
B5-480VAC-LF	480 VAC Buses	AB/371'	Site Specific	5-10	26	
B6-480VAC-LF		AB/371'	Site Specific	5-10	26	
B55-480VAC-LF		AB/371'	Site Specific	5-10	26	
B56-480VAC-LF		AB/371'	Site Specific	5-10	26	
B53-480VAC-LF		AB/371'	Site Specific	5-10	26	
B51-480VAC-LF		AB/371'	Site Specific	5-10	26	
B52-480VAC-LF		AB/371'	Site Specific	5-10	26	
B63-480VAC-LF		AB/386'	Site Specific	5-10	22	
B61-480VAC-LF		AB/386'	Site Specific	5-10	22	
B62-480VAC-LF		AB/386'	Site Specific	5-10	22	
RA1-125VDC-LF	125 VDC Buses	AB/371'	10	5-10	26	
RA2-125VDC-LF		AB/371'	10	5-10	26	
D11-125VDC-LF		AB/371'	10	5-10	26	
D21-125VDC-LF		AB/371'	10	5-10	26	
D02-125VDC-LF		AB/371'	10	5-10	26	
D01-125VDC-LF		AB/371'	10	5-10	26	
LF-DC-RS1		AB/371'	10	5-10	26	
LF-DC-RS2		AB/371'	10	5-10	26	
DG1-GEN-LF	4KV Diesel Generators	AB/371'	14	22	28	
DG2-GEN-LF		AB/371'	14	22	28	
LF-DC-Y11	Inverters	AB/371'	6	5-10	26	
LF-DC-Y22		AB/371'	6	5-10	26	
LF-DC-Y11AC	Inverter AC Power Circuit	AB/371'	3	5-10	26	
LF-DC-Y22AC		AB/371'	3	5-10	26	

Table 5.4 Arkansas Nuclear One Seismic Failure Basic Event Data (Cont.)

<u>Basic Event</u>	<u>Description</u>	<u>Location/Elev.</u>	<u>Fragility Category</u>	<u>Estimated Frequency</u>	<u>Response Number</u>	<u>Comment</u>
LF-DC-Y11DC	Inverter DC	AB/371'	3	5-10	26	
LF-DC-Y22DC	Power Circuit	AB/371'	3	5-10	26	
LF-DC-RS1ALTAC	Alternate AC Power	AB/371'	3	5-10	26	
LF-DC-RS/ALTAC	to RS Bus	AB/371'	3	5-10	26	
BATT07-ZBT-LF	125 VDC	AB/371'	4	ZPA	28	
BATT06-ZBT-LF	Station Batteries	AB/371'	4	ZPA	28	
D03-CHARGER-LF	Chargers	AB/371'	3	5-10	26	
D04-CHARGER-LF		AB/371'	3	5-10	26	
LOSS-OSP	Ceramic Insulators	Yard/353'	1	ZPA	40	
ACT-EFAA-LF		AB/386'	3	5-10	22	
ACT-EFBB-LF	EFW I&C	AB/386'	3	5-10	22	
ACT-EFXA-LF	Logic Trains	AB/386'	3	5-10	22	
ACT-EFXB-LF		AB/386'	3	5-10	22	
LF-REF-RM95	Battery Rooms	AB/371'	16	7	26	
LF-REF-RM109	Room Cooling Refrigeration Units	AB/371'	16	7	26	

Table 5.4 Arkansas Nuclear One Seismic Failure Basic Event Data (Cont.)

Basic Event	Description	Location/Elev.	Fragility Category	Estimated Frequency	Response Number	Comment
LF-ECS-A1	Ventilation System Segment Failures	AB/371'	22	ZPA	28	
LF-ECS-B1		AB/371'	22	ZPA	28	
LF-ECS-A2		AB/371'	22	ZPA	28	
LF-ECS-B2		AB/371'	22	ZPA	28	
LF-ECS-A4		AB/371'	22	ZPA	28	
LF-ECS-B4		AB/371'	22	ZPA	28	
LF-ECS-A5		AB/371'	22	ZPA	28	
LF-ECS-B5		AB/371'	22	ZPA	28	
LF-ECS-A6		AB/371'	22	ZPA	28	
LF-ECS-B6		AB/371'	22	ZPA	28	
LF-ESF-T1	Digital Logic Circuits	AB/386'	3	5-10	22	
LF-ESF-T2		AB/386'	3	5-10	22	
LF-ESF-T3		AB/386'	3	5-10	22	
LF-ESF-T4		AB/386'	3	5-10	22	
LF-ESF-T5		AB/386'	3	5-10	22	
LF-ESF-T6		AB/386'	3	5-10	22	
LF-ESF-T7		AB/386'	3	5-10	22	
LF-ESF-T8		AB/386'	3	5-10	22	
LF-ESF-TRS1	Channel	AB/386'	3	5-10	22	
LF-ESF-TRS2	Power Supplies	AB/386'	3	5-10	22	
ESFA110X-LOG-LF	Analog	AB/386'	3	5-10	22	
ESFA210X-LOG-LF	Logic Units	AB/386'	3	5-10	22	

Table 5.4 Arkansas Nuclear One Seismic Failure Basic Event Data (Cont.)

Basic Event	Description	Location/Elev.	Fragility Category	Estimated Frequency	Response Number	Comment
ESFA310X-LOG-LF		AB/386'	3	5-10	22	
ESFA116X-LOG-LF		AB/386'	3	5-10	22	
ESFA216X-LOG-LF		AB/386'	3	5-10	22	
ESFA316X-LOG-LF		AB/386'	3	5-10	22	
ESFA122-LOG-LF	Analog Logic Units	AB/386'	3	5-10	22	
ESFA222-LOG-LF		AB/386'	3	5-10	22	
ESFA322-LOG-LF		AB/386'	3	5-10	22	
ESFA123-LOG-LF		AB/386'	3	5-10	22	
ESFA223-LOG-LF		AB/386'	3	5-10	22	
ESFA323-LOG-LF		AB/386'	3	5-10	22	
LF-ESF-TA01	Analog Signal Path with Bistable	AB/386'	3	5-10	22	
LF-ESF-TB01		AB/386'	3	5-10	22	
LF-ESF-TC01		AB/386'	3	5-10	22	
LF-ESF-TA120	Analog Bistables	AB/386'	3	5-10	22	
LF-ESF-TA220		AB/386'	3	5-10	22	
LF-ESF-TA320		AB/386'	3	5-10	22	
LF-ESF-TA119		AB/386'	3	5-10	22	
LF-ESF-TA219		AB/386'	3	5-10	22	
LF-ESF-TA319		AB/386'	3	5-10	22	
LF-ESF-TA04	Analog Signal Paths	AB/386'	3	5-10	22	
LF-ESF-TB04		AB/386'	3	5-10	22	
LF-ESF-TC04		AB/386'	3	5-10	22	
LF-SWS-VCH4A	SW Chill Water	SW Pump Bldg./353'	28	5	37	
LF-SWS-VCH4B	Cooling Units	SW Pump Bldg./353'	28	5	37	

Table 5.4 Arkansas Nuclear One Seismic Failure Basic Event Data (Cont.)

<u>Basic Event</u>	<u>Description</u>	<u>Location/Elev.</u>	<u>Fragility Category</u>	<u>Estimated Frequency</u>	<u>Response Number</u>	<u>Comment</u>
CST-TNK-LF	Condensate Storage Tank	Yard/353'	Site Specific	ZPA	40	
BWST-TNK-LF	Borated Water Storage Tank	Yard/353'	Site Specific	20	39	
HPRE35AX-HTX-LFB	RHR Heat Exchangers	AB/317'	25	ZPA	36	
HPRE35BX-HTX-LFB		AB/317'	25	ZPA	36	
LF-RBC-SEGCB	RBCS Segment with Fan	RB/401.5'	28	5	14	
LF-RBC-SEGAA		RB/401.5'	28	5	14	
LF-RBC-SEGBA		RB/401.5'	28	5	14	
LF-RBC-SEGDB		RB/401.5'	28	5	14	
LF-RBC-SWL1	RBCS Segment with Heat Exchanger	RB/401.5'	24	20	15	
LF-RBC-SWL2		RB/401.5'	24	20	15	

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responses were also derived from SSMRP response calculations for the Zion plant. These rules are summarized in Table 5.5. Applications of these rules to ANO-1 results in the (symmetric) response correlation matrix given in Table 5.6. By use of this table, the correlation between any pair of responses listed in Table 5.3 can be determined. These correlations are needed in evaluating the probabilities for cut sets involving dependent failures, as described in Chapter 6.

Table 5.5

Rules for Assigning Response Correlation $\rho_{R_1 R_2}$

1. Components on the same floor slab, and sensitive to the same spectral frequency range (i.e., ZPA, 5-10 Hz, or 10-15 Hz) will be assigned response correlation = 1.0.
 2. Components on the same floor slab, sensitive to different ranges of spectral acceleration will be assigned response correlation = 0.5.
 3. Components on different floor slabs (but in the same building) and sensitive to the same spectral frequency range (ZPA, 5-10 Hz or 10-15 Hz) will be assigned response correlation = 0.75.
 4. Components on the ground surface (outside tanks, etc.) shall be treated as if they were on the grade floor of an adjacent building.
 5. "Ganged" valve configurations (either parallel or series) will have response correlation = 1.0.
 6. All other configurations will have response correlation equal to zero.
-

Table 5.6 Correlation Coefficients Between Responses -
ANO-1 (Matrix is symmetric. Above diagonal,
all blanks are zeros.)

	Response Number									
	14	15	22	26	28	34	36	37	39	40
14	1.0	0.5								
15		1.0								
22			1.0	0.75		0.75				
26				1.0	0.5	0.75				
28					1.0		0.75			
34						1.0	0.5			
36							1.0			
37								1.0	0.5	0.5
39									1.0	0.5
40										1.0

6.0 SEISMIC FRAGILITIES

Component failure is taken as either loss of pressure boundary integrity or loss of operability. Failure (fragility) is characterized by a cumulative distribution function which describes the probability that failure has occurred given a value of loading. Loading may be described by local spectral acceleration or moment, depending on the component failure mode. The fragilities are related to the local response to permit an accurate assessment of the effect of earthquake induced correlation in the evaluation of the accident sequences.

6.1 Generic Fragilities

A generic data base of fragility functions for seismically-induced failures was developed in the SSMRP (Reference 4). As a first step, all components were grouped into generic categories. For example, all motor operated valves located on piping with diameters between 2-1/2 and 8 inches were placed into a single generic category, and similarly, all motor control centers were placed into another generic category. The generic categories are shown on Table 6.1.

Fragility functions for the generic categories were developed based on a combination of experimental data, design analysis reports, and an extensive expert opinion survey. The experimental data utilized in developing fragility curves were obtained from the results of component manufacturer's qualification tests, independent testing lab failure data and data obtained from the extensive U.S. Corps of Engineers SAFEGUARD Subsystem Hardness Assurance Program. These data were critically examined for applicability and then statistically combined with the expert opinion survey data to produce the final fragility curves for the generic component categories (Reference 4).

Finally, since this fragility data base was assembled in the 1980-1982 time frame, a review of site-specific fragilities developed during the past four years was performed by R. Campbell of NTS Structural Mechanics Associates, Inc. (Attachment C.2). This review addressed the seismic fragilities developed for the PRAs of 19 different power plants. As a result of this review, the median fragility levels of 5 of the generic categories in the original SSMRP data base (Reference 4) were revised. (However, no substantial differences were found from the fragility values used in the SSMRP Zion analysis.)

Based on the above, a final set of generic fragility values was chosen for use in TAP A-45. The median failure accelerations and both random and systematic uncertainties (expressed as standard deviations of the logarithms of the fragility) are listed in Table 6.2. Values of the failure probabilities of the generic component categories, as a function of peak ground

Table 6.1 Fragility Categories

Frag. Cat.	Component Class	Typical Components	Fragility (Hz)
1	LOSP	Ceramic Insulators	ZPA
2	Relays		5-10
3	Circuit Breakers		5-10
4	Batteries		ZPA
5	Battery Racks		ZPA
6	Inverters		5-10
7	Transformers	4KV to 480V and 480 to 120V	10
8	Motor Control Centers	Control for ESF Pumps and Valves	5-10
9	Aux. Relay Cabinets		5-10
10	Switchgear (Inc. Transformers, Buses & Breakers)	4160V and 480V	5-10
11	Cable Trays		ZPA
12	Control Panels and Racks	RPS Process Control	5-10
13	Local Instruments	Misc. Pressure & Temperature Sensors	5-35
14	Diesel Generators	4160 AC Emergency Power Units	22
15	Horizontal Motors	Motor-Generator Sets	ZPA
16	Motor-Driven Pumps and Compressors	AFWS, RHR, SIS, Charging Pumps Lube Oil Pumps, Diesel Starting Compressors	7
17	Large Vertical Centrifugal Pumps (Motor-Driven)	Service Water Pumps	5
18	Large Motor-Operated Valves (> 10")		ZPA
19	Small Motor-Operated Valves (< 10")		ZPA
20	Large Pneumatic/Hydraulic Valves	Includes MSIV, ADS, and PORV	ZPA
21	Large Check and Relief Valves		ZPA
22	Miscellaneous Small Valves (< 8")		ZPA
23	Large Horizontal Vessels & Heat Exchangers	Pressurizer Relief Tank, CCW Heat Exchangers	ZPA
24	Small to Medium Heat Exchangers & Vessels	Boron Injection Tank	20
25	Large Vertical Storage Vessels w/Formed Heads	RHR Heat Exchanger, Accumulator Tank	ZPA
26	Large Vertical Flat-Bottomed Storage Tanks	CST, RWST	
27	Master Piping Fragility		
28	Air Handling Units	Containment Fan Coolers	5

Table 6.2 Generic Component Fragilities, in units of Gravity (g).
Except for Piping, which is in units of Moment (in.lb.)

Category	Generic Component	Median*	β_R	β_U
1	Ceramic insulators	0.25	0.25	0.25
2	Relays	4.00	0.48	0.75
3	Circuit breakers	7.63	0.48	0.74
4	Batteries	0.80	0.40	0.39
5	Battery racks	2.29	0.31	0.39
6	Inverters	2.00	0.26	0.35
7	Dry transformers	8.80	0.28	0.30
8	Motor control centers	7.63	0.48	0.74
9	Auxiliary relay cabinets	7.63	0.48	0.74
10	Switchgear	6.43	0.29	0.66
11	Cable trays	2.23	0.34	0.19
12	Control panels and racks	11.50	0.48	0.74
13	Local instruments	7.68	0.20	0.35
14	Diesel generators	1.00	0.25	0.31
15	Horizontal motors	12.10	0.27	0.31
16	Motor-driven pumps and compressors	2.80	0.25	0.27
17	Large vertical centrifugal pumps	2.21	0.22	0.32
18	Large motor-operated valves (> 10 in.)	6.50	0.26	0.60
19	Small motor-operated valves (< 10 in.)	4.83	0.26	0.35
20	Large pneumatic/hydraulic valves	6.50	0.26	0.35
21	Large relief, manual, and check valves	8.90	0.20	0.35
22	Misc. small valves	12.50	0.33	0.43
23	Large horizontal vessels and heat exchangers	3.00	0.30	0.53
24	Small to medium vessels and heat exchangers	1.84	0.25	0.45
25	Large vertical vessels with formed heads	1.46	0.20	0.35
26	Large vertical tanks with flat bottoms	0.45	0.25	0.29
27	Piping (master fragility)	2.44 x 10 ⁶	0.18	0.33
28	Air handling units	6.90	0.27	0.61

β_R - random uncertainty

β_U = systematic uncertainty

* All medians in terms of spectral acceleration were developed for 5% damping.

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acceleration level and for various floor amplification factors, are given in Attachment C.1. Note that these apply specifically to a site with an SSE of 0.20g. Note also that these failure probabilities have been computed based on random uncertainties in both fragility and response, and the response uncertainties were assigned as described in Section 5.5. The failure probabilities in Attachment C.1 were used for all components except those for which site-specific fragilities were developed.

Fragilities for electrical components represent a special problem in that there is a wide variety of electrical gear found within a plant. Typically, all this gear is enclosed in switchgear or motor control centers. The two lowest failure modes in the fragility data base are relay chatter and inadvertent trip of circuit breakers. Virtually all the electrical switching and control equipment in a nuclear power plant includes these two types of components, so these two failure modes are used as generic failure modes for all electrical gear. Relay chatter is the lowest failure mode and, if included in a risk analysis, would be the dominant failure. Because, in most cases, circuits are protected by time delay circuits and because, in most cases, chatter of relays would not cause a change in the state of the system being controlled, we chose not to include relay chatter as a failure mode for electrical gear but rather included only circuit breaker trip.

6.2 Site-Specific Fragilities

During the plant visit (April 86), the following components were identified as requiring plant-specific fragility derivations:

- 1) BWST
- 2) 4160 VAC Switchgear
- 3) 480 VAC Switchgear

The BWST tank was identified both because of its height-to-diameter ratio as well as due to the time period during which it was designed and installed. No drawings were provided to allow a site-specific analysis of its fragility. However, considering the time period in which it was designed and installed (prior to 1974), the BWST was undoubtedly designed according to the procedures in TID-7024 which can lead to a non-conservative design in some cases. Based on past experience, a median fragility level of 0.40g was assigned to the BWST. If site-specific analysis showed a lower value, some additional impact on the seismic risk values would be expected. The 4160 VAC switchgear and 480 VAC switchgear were observed during the plant visit not to be anchored. Based on these observations, site-specific fragilities were developed for all the 4160 VAC and 480 VAC switchgear. The resulting site-specific fragilities are summarized on Table 6.3.

Table 6.3 Summary of ANO-1 Site-Specific
Fragility Functions

<u>Component</u>	<u>Failure Mode</u>	<u>Median Base Acceleration at Failure</u>	<u>β Random Uncertainty</u>
4160 VAC Switchgear	Sliding & Tipping Due to Lack of Anchorage	0.324g	0.4
480 VAC Switchgear	Sliding & Tipping Due to Lack of Anchorage	0.324g	0.4

It should be pointed out that all the components above were found to have median failure acceleration levels above the SSE. However, they did have less margin of safety above the SSE than the other components examined during the plant visit, and hence were anticipated to be significant contributors to the accident sequence probabilities.

7.0 CORE DAMAGE AND RISK COMPUTATIONS

7.1 Event Trees

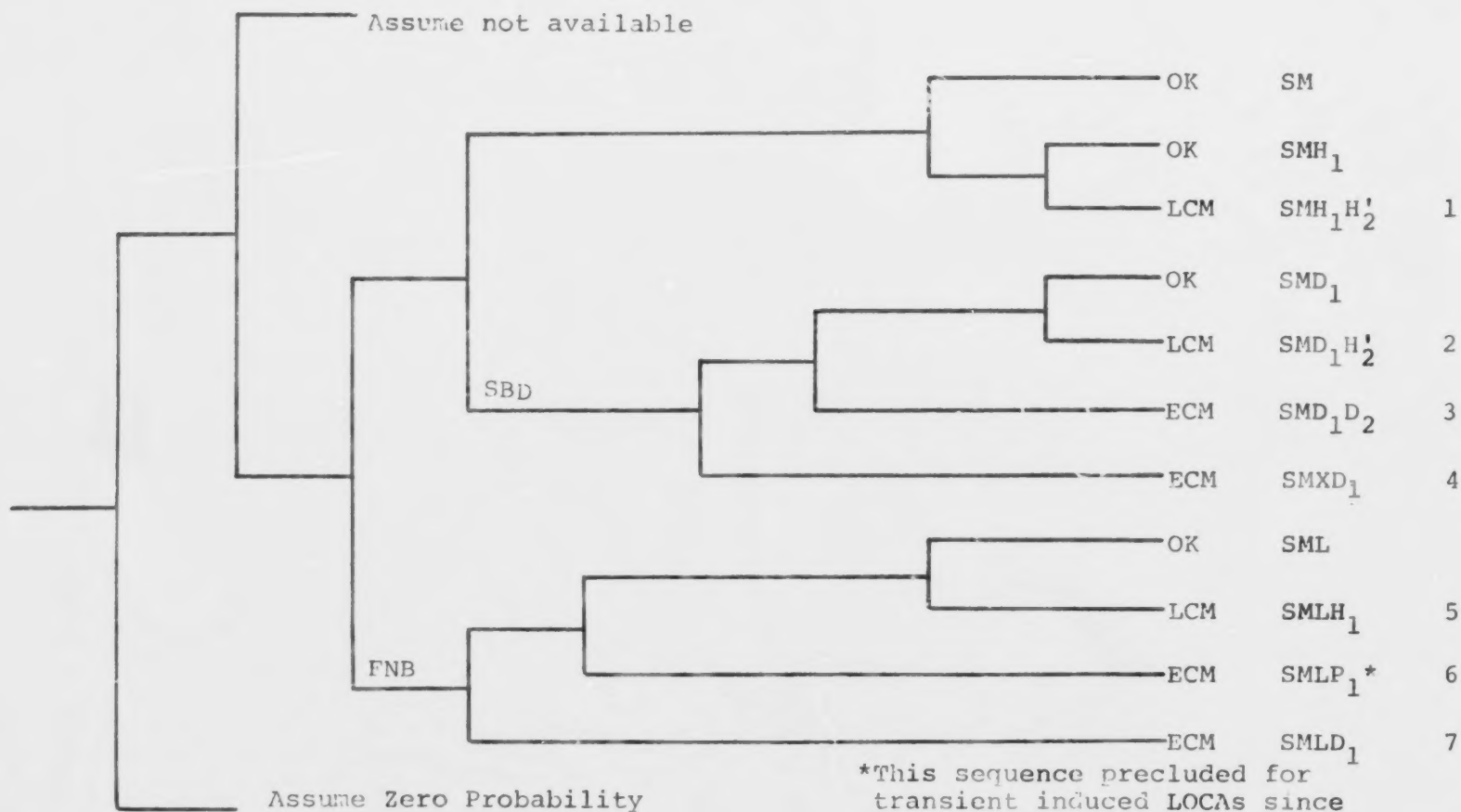
The event trees used for the seismic analysis were taken from those used for the internal events analysis, and are shown in Figures 7.1 through 7.3. Note that failure of the reactor protection system is not in the scope of the TAP A-45 analyses, so no branches corresponding to failure to SCRAM are shown on any of the event trees.

Seven small and seven small-small LOCA sequences leading to core damage are identified in Figures 7.1a and 7.1b. The first four sequences involve cases in which the emergency feedwater system (L) succeeds, and thus secondary heat removal is assured, and the remaining challenge to the safety systems is to cool the core while depressurizing the primary system. This can be accomplished using the HPI system (with pressure relief via the pressurizer safety relief valves) or, if the HPI system is not available, using secondary blowdown (event X). In this latter case, rapid depressurization of the primary system is accomplished using the pressurizer sprays in conjunction with either both secondary relief (atmospheric dump) valves or the steam dump valves to the main condenser. Following successful secondary blowdown, the low pressure system is used in both injection and recirculation phases to achieve hot shutdown conditions. Note that the RHR heat exchangers are not needed in these sequences (as denoted by the prime on event H₂) since the emergency feedwater system provides the necessary secondary heat removal.

Sequences 5 through 7 are cases in which the emergency feedwater system has failed. For these cases, a "feed and bleed" mode of cooling must be used in which coolant is injected into the primary system using the HPI system, and heat removal is accomplished by releasing steam (or water) from the pressurizer power-operated relief valves. Following the injection phase, the high pressure recirculation system (HPR) is used to continue the "feed and bleed" cooling mode.

The two transient event trees are shown in Figures 7.2 and 7.3. For the T₂ transient, the main feedwater system is assumed not available. In each case the emergency feedwater system is called upon to provide secondary cooling. Opening and closing of the pressurizer safety relief valves is required to relieve primary system overpressure. Failure of a relief valve to reclose results in a leakage equivalent to a small LOCA. If the emergency feedwater system fails, "feed and bleed" cooling is required to prevent core damage, as was the case for the small LOCA sequences. The important accident sequences are found to be those resulting from failure of feed and bleed, in this case sequences T₂-2 and T₃-2.

a	S ₃	K	M	L	E	P ₁	X	D ₂	H ₁	H' ₂			
b	S ₂	K	M	L	D ₁	P ₁	X	D ₂	H ₁	H' ₂	Core Status	Accident Sequence	Sequence Number
	LOCA	RPS	PCS	EFWS	HPIS	SRV & ERV	SBD	LPIS	HPRS	LPRS			



*This sequence precluded for transient induced LOCAs since P was by definition successful in order that Q fail

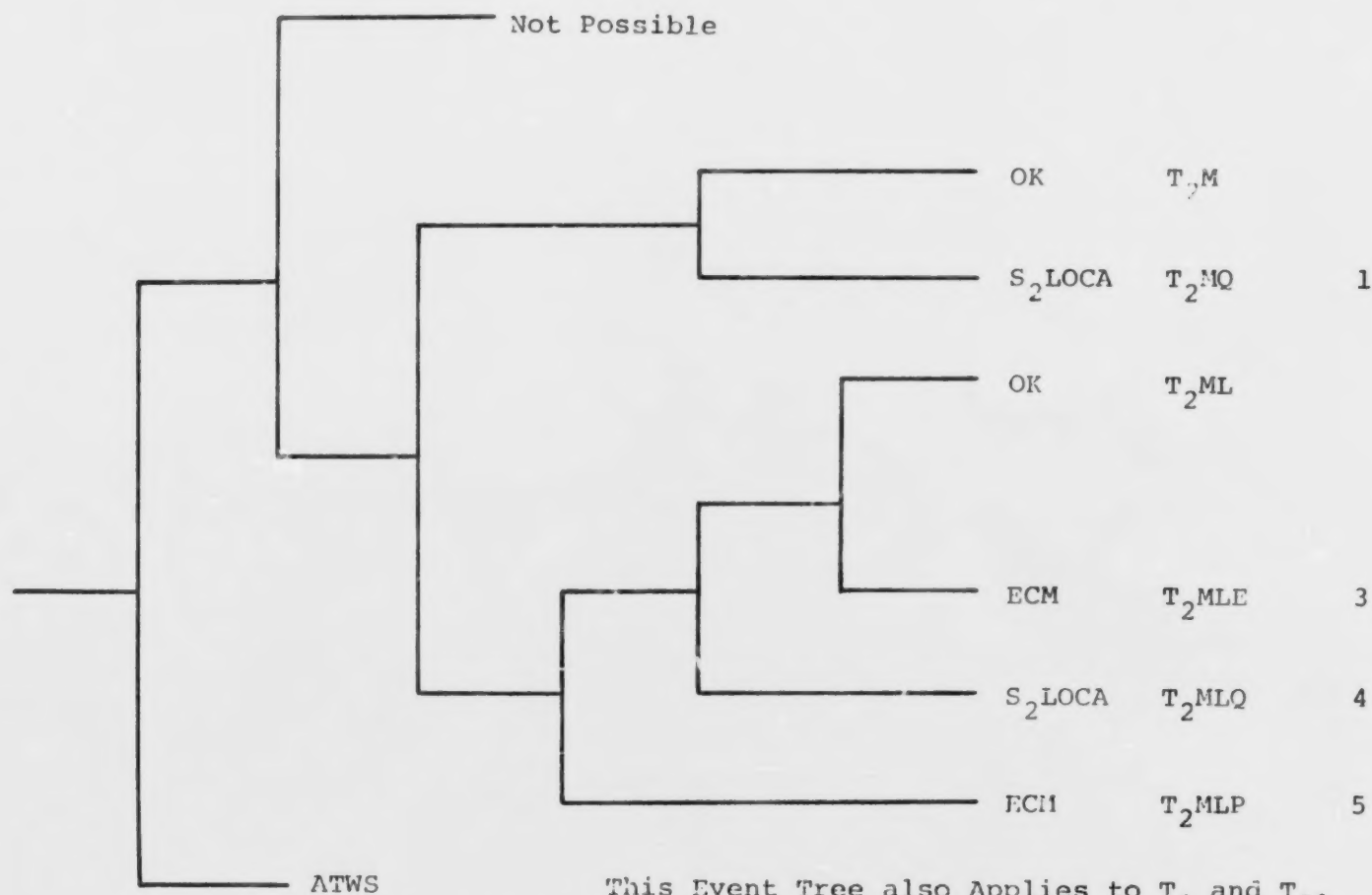
Figure 7.1 Small LOCA (S₂) Event Tree

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T ₂	K	M	L	P	Q	E
LOF	RPS	PCS	EFWS	SRVO	SRVR	HPIS

CORE
STATUSACCIDENT
SEQUENCESEQUENCE
NUMBERThis Event Tree also Applies to T₄ and T₅.Figure 7.2 Loss of Feedwater Transient (T₂) Event Tree

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T ₃	K	M	L	P	Q	E
AOT	RPS	PCS	EFWS	SRVO	SRVR	HPIS

CORE
STATUS ACCIDENT
SEQUENCE SEQUENCE
NUMBER

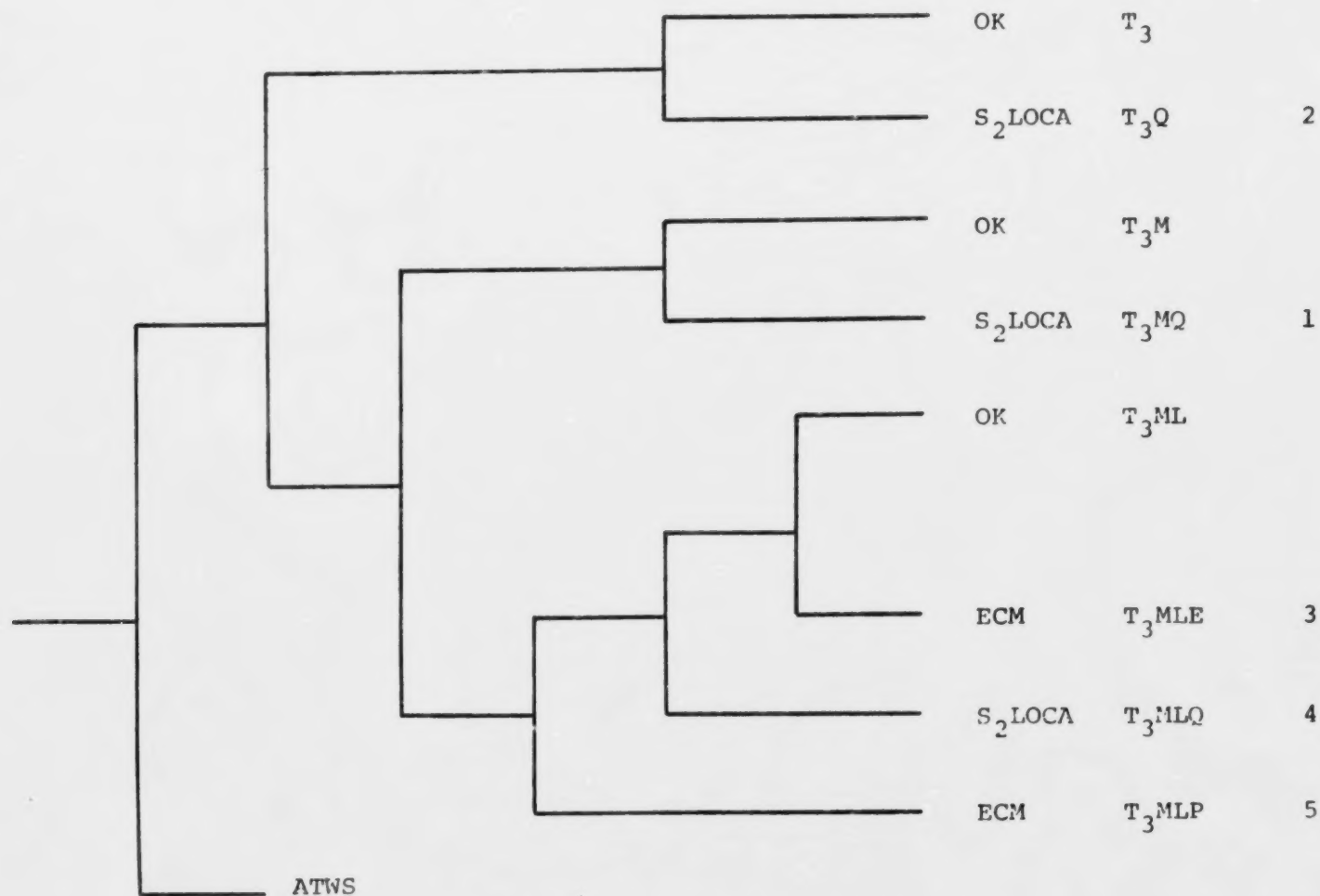


Figure 7.3 All Other Transients (T₃) Event Tree

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7.2 Initiating Events

Four initiating events are being considered:

- | | | |
|-----|------------------|--|
| (a) | Small-Small LOCA | (pipe break with ID < 1.2 inches) |
| (b) | Small LOCA | (pipe break with ID between 1.2 and 1.66 inches) |
| (c) | Transient Type 2 | (PCS initially inoperative) |
| (d) | Transient Type 3 | (PCS initially available) |

The frequency of the SLOCA and SSLOCA initiating events are taken from the calculations of pipe failure frequencies computed as part of the SSMRP. In this program, multi-support time history analysis of some 22 piping systems were made, and failure probabilities were computed for the weakest element (branch, tree, etc.) in each run of piping between valves and/or supports. These calculations included pipes from 3 inches to 29 inches and later analysis showed that the failure probabilities were not correlated with pipe size. From this data set, the probabilities of various sized LOCAs were computed, as described in Reference 3, at six levels of peak ground acceleration. The probability of SSLOCA (given that a small LOCA has not occurred) is:

$$P(SSLOCA) = P(SSLOCA) \overline{P(SLOCA)}$$

The frequency of the Type 2 transient initiating event (in which loss of offsite power is implied) is computed based on the failure of ceramic insulators. These have a relatively low median failure level, and are the dominant cause of seismically-induced loss of offsite power. The probability of the T₂ transient (given that a small LOCA and a small-small LOCA have not occurred) is:

$$P(T_2) = P(\text{Ceramic Insulators}) \overline{P(SLOCA)} \overline{P(SSLOCA)}$$

Finally, the Type 3 initiating event probability is computed from the condition that the sum of the initiating event probabilities considered must be unity. The hypothesis here is that, given an earthquake of reasonable size, at least one of the initiating events will occur. Hence, we compute the probability of the Type 3 transient initiating event from:

$$P(T_3) = 1 - P(T_2) - P(SLOCA) - P(SSLOCA)$$

The probabilities of the all four initiating events at the four earthquake levels under consideration are given in Table 7.1. At the lower earthquake levels, the probability of SSLOCA and SLOCA are small and the transient events dominate, while the reverse is true at the higher earthquake levels.

Table 7.1 Initiating Event Probabilities

	<u>0.1g</u>	<u>0.2g</u>	<u>0.4g</u>	<u>0.6g</u>
SSLOCA	9.0E-4	4.3E-3	8.0E-2	2.6E-1
SLOCA	1.0E-6	1.5E-3	3.8E-2	1.5E-1
T ₂	4.8E-3	2.6E-1	8.8E-1	5.9E-1
T ₃	9.9E-1	7.4E-1	1.0E-6	1.0E-6

7.3 Base Case Accident Sequence Probabilities

This section presents the results of the base case seismic risk analysis for the ANO-1 nuclear power plant. The base case is our best estimate of the current configuration of the plant and its emergency procedures. In particular, the seismic component failure probabilities were taken from the generic data base (Appendix C.1) with the exception of the following site-specific component failure modes:

<u>Component</u>	<u>Best Estimate Failure Mode</u>
4160 VAC Switchgear	Sliding and tipping due to lack of floor anchorage
480 VAC Switchgear	Sliding and tipping due to lack of floor anchorage

Table 7.2 lists the top eight accident sequence probabilities at each earthquake level. These values include the initiating event probabilities (Table 7.1) and the earthquake occurrence probabilities (as given on Table 4.3). The total core melt probability (the sum of all accident sequence probabilities summed over all earthquake levels) is seen to be 7.3E-5 per year. The last column on the right hand side gives the contribution to the probability of core melt from each accident sequence. It can be seen that the dominant accident sequences are (in order of importance):

T₂(2) (70%)
 SSLOCA(3) (10%)
 T₃(3) (10%)
 SSLOCA(7) (4%)

Table 7.2 Base Case Accident Sequence
Probabilities per Year - ANO-1

Sequence	Earthquake Magnitude				Total
	0.100- 0.200g	0.200- 0.400g	0.400- 0.600g	Above 0.600g	
SLOCA(3)	1.14E-7	1.69E-6	7.31E-7	4.32E-9	2.54E-6
SLOCA(4)	3.73E-9	4.17E-8	2.46E-8	1.36E-10	7.02E-8
SLOCA(7)	1.52E-8	5.07E-7	1.02E-6	4.40E-10	1.54E-6
T2(2)	6.28E-6	3.13E-5	1.30E-5	3.04E-7	5.08E-5
T3(2)	5.88E-6	1.24E-6	0.0	1.38E-7	7.26E-6
SSLOCA(3)	1.95E-6	4.14E-6	1.40E-6	6.24E-8	7.54E-6
SSLOCA(4)	4.85E-8	9.75E-8	4.60E-8	1.38E-9	1.93E-7
SSLOCA(7)	1.39E-7	1.06E-6	1.85E-6	4.80E-9	3.05E-6
TOTAL	1.44E-5	4.00E-5	1.80E-5	5.15E-7	7.30E-5

Taken together, these four sequences constitute 94% of the total core melt probability for the base case.

The largest contributor to the core damage is the transient sequence

$$T_2(2) = T_2 \text{ ML}\bar{P}\bar{Q}E$$

in which the emergency feedwater system (L) fails and the PCS(M) fails due to the initiator. The important cut sets in this sequence are shown in Table 7.3. Note that the same cut sets apply to the third most important sequence,

$$T_3(2) = T_3 \text{ ML}\bar{P}\bar{Q}E$$

the only difference being that, in the $T_3(2)$ sequence, loss of the main feedwater system is not given, but arises due to LOSP.

The second greatest contributor to core damage is the sequence SSLOCA(3), given by

$$\text{SSLOCA}(3) = S_3 \text{ M}\bar{L}\bar{E}\bar{X}D_2$$

As shown in Table 7.4, this is dominated by the cut sets involving failure of electrical busses. Other cut sets are at least an order of magnitude smaller. SLOCA(3) has the same dominant cut sets.

The fourth greatest contributor is the sequence SSLOCA(7). As shown in Table 7.5, the dominant cutsets are similar to $T_2(2)$. SLOCA(7) has the same dominant cut sets..

For completeness, the dominant cut sets in SLOCA(4) and SSLOCA(4) are shown in Table 7.6.

No small LOCA sequences appear in the top 94% of all sequences for two main reasons. First, the initiator value at all seismic levels is always less than the corresponding value of small-small LOCA initiator and their dominant cut sets are identical. Secondly, transient initiators are dominant at lower seismic levels where a majority of the risk profile occurs. The BWST does not appear as a dominant basic event in the base case analysis because of the masking effect due to the 4160 VAC and 480 VAC switchgear having such a high failure probability.

7.4 Release Category Assignment

Figures 7.4 and 7.5 show the containment event tree used both for the internal event analysis and the seismic analysis. The LOCA containment event tree involves success or failure of the

Table 7.3 Dominant Cut Sets in Sequence $T_2(2)$ and $T_3(2)$

	Earthquake Level				Containment Tree Branch
	0.1g	0.2g	0.4g	0.6g	
A3-4160VAC-LF * B6-480VAC-LF * EFWOP7AX-PTD-LF	1.25E-2	3.01E-2	3.57E-2	4.90E-2	Z_T
A3-4160VAC-LF * A4-4160VAC-LF * EFWOP7AX-PTD-LF	1.25E-2	3.01E-2	3.57E-2	4.90E-2	Z_T
A3-4160VAC-LF * B61-480VAC-LF * EFWOP7AX-PTD-LF	1.06E-2	2.97E-2	3.57E-2	4.90E-2	$\bar{Z}_T^C$
A3-4160VAC-LF * B62-480VAC-LF * EFWOP7AX-PTD-LF	1.06E-2	2.97E-2	3.57E-2	4.90E-2	$\bar{Z}_T^{\bar{C}}$
BATT-CM	5.20E-4	9.50E-4	3.10E-2	1.60E-1	Z_T
B51-480VAC-LF * B61-480VAC-LF * CST-TNK-LF	*	*	9.32E-3	9.50E-2	$\bar{Z}_T^C$
B51-480VAC-LF * B6-480VAC-LF * CST-TNK-LF	*	*	9.32E-3	9.50E-2	$\bar{Z}_T^C$
B5-480VAC-LF * B61-480VAC-LF * CST-TNK-LF	*	*	9.32E-3	9.50E-2	$\bar{Z}_T^C$
B5-480VAC-LF * B6-480VAC-LF * CST-TNK-LF	*	*	9.32E-3	9.50E-2	Z_T
A3-4160VAC-LF * B61-480VAC-LF * CST-TNK-LF	*	*	9.32E-3	9.50E-2	$\bar{Z}_T^C$
A3-4160VAC-LF * B6-480VAC-LF * CST-TNK-LF	*	*	9.32E-3	9.50E-2	Z_T
B51-480VAC-LF * A4-4160VAC-LF * CST-TNK-LF	*	*	9.32E-3	9.50E-2	$\bar{Z}_T^C$

* Denotes negligible

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Table 7.2 Dominant Cut Sets in Sequence $T_2(2)$ and $T_3(2)$ (continued)

	Earthquake Level				Containment Tree Branch
	0.1g	0.2g	0.4g	0.6g	
B51-480VAC-LF * B62-480VAC-LF * CST-TNK-LF	*	*	9.32E-3	9.50E-2	$\bar{Z}_T \bar{C}$
B5-480VAC-LF * A4-4160VAC-LF * CST-TNK-LF	*	*	9.32E-3	9.50E-2	Z_T
B5-480VAC-LF * B62-480VAC-LF * CST-TNK-LF	*	*	9.32E-3	9.50E-2	$\bar{Z}_T \bar{C}$
A3-4160VAC-LF * A4-4160VAC-LF * CST-TNK-LF	*	*	9.32E-3	9.50E-2	Z_T
A3-4160VAC-LF * B62-480VAC-LF * CST-TNK-LF	*	*	9.32E-3	9.50E-2	$\bar{Z}_T \bar{C}$
BWST-TNK-LF * CST-TNK-LF	*	*	6.73E-3	9.00E-2	$\bar{Z}_T C$
A3-4160VAC-LF * B61-480VAC-LF * EFW2802B-VOC-LF	9.14E-6	7.29E-5	1.09E-2	8.80E-2	$\bar{Z}_T C$
A3-4160VAC-LF * B6-480VAC-LF * EFW2802B-VOC-LF	9.14E-6	7.29E-5	1.09E-2	8.80E-2	Z_T
A3-4160VAC-LF * A4-4160VAC-LF * EFW2802B-VOC-LF	9.14E-6	7.29E-5	1.09E-2	8.80E-2	Z_T
A3-4160VAC-LF * B62-480VAC-LF * EFW2802B-VOC-LF	9.14E-6	7.29E-5	1.09E-2	8.80E-2	$\bar{Z}_T \bar{C}$
A3-4160VAC-LF * BWST-TNK-LF * EFW2802B-VOC-LF	*	9.27E-6	2.56E-2	8.32E-2	$\bar{Z}_T C$

* Denotes negligible

Table 7.4 Dominant Cut Sets in Sequence $S_2(3)$ and $S_3(3)$

	Earthquake Level				Containment Tree Branch
	0.1g	0.2g	0.4g	0.6g	
B5-480VAC-LF * B6-480VAC-LF	3.58E-1	8.61E-1	9.93E-1	9.99E-1	$Z_L C$
A3-4160VAC-LF * B6-480VAC-LF	3.58E-1	8.61E-1	9.93E-1	9.99E-1	$Z_L C$
B51-480VAC-LF * B6-480VAC-LF	3.58E-1	8.61E-1	9.93E-1	9.99E-1	$\bar{Z}_L C$
B5-480VAC-LF * A4-4160VAC-LF	3.58E-1	8.61E-1	9.93E-1	9.99E-1	$Z_L C$
A3-4160VAC-LF * A4-4160VAC-LF	3.58E-1	8.61E-1	9.93E-1	9.99E-1	$Z_L C$
B51-480VAC-LF * A4-4160VAC-LF	3.58E-1	8.61E-1	9.93E-1	9.99E-1	$\bar{Z}_L C$
B5-480VAC-LF * B61-480VAC-LF	3.28E-1	8.50E-1	9.92E-1	9.99E-1	$\bar{Z}_L C$
A3-4160VAC-LF * B61-480VAC-LF	3.28E-1	8.50E-1	9.92E-1	9.99E-1	$\bar{Z}_L C$
B51-480VAC-LF * B61-480VAC-LF	3.28E-1	8.50E-1	9.92E-1	9.99E-1	$\bar{Z}_L C$
A3-4160VAC-LF * B62-480VAC-LF	3.28E-1	8.50E-1	9.92E-1	9.99E-1	$Z_L \bar{C}\bar{F}$

Table 7.5 Dominant Cut Sets in Sequence $S_2(7)$ and $S_3(7)$

	Earthquake Level				Containment Tree Branch
	0.1g	0.2g	0.4g	0.6 g	
A3-4160VAC-LF * B6-480VAC-LF * EFWOP7AX-PTD-LF	1.25E-2	3.01E-2	3.57E-2	4.90E-2	$Z_L C$
A3-4160VAC-LF * A4-4160VAC-LF * EFWOP7AX-PTD-LF	1.25E-2	3.01E-2	3.57E-2	4.90E-2	$Z_L C$
A3-4160VAC-LF * B61-480VAC-LF * EFWOP7AX-PTD-LF	1.06E-2	2.97E-2	3.57E-2	4.90E-2	$\bar{Z}_L C$
A3-4160VAC-LF * B62-480VAC-LF * EFWOP7AX-PTD-LF	1.06E-2	2.97E-2	3.57E-2	4.90E-2	$Z_L \bar{C} \bar{F}$
BATT-CM	5.20E-4	9.50E-4	3.10E-2	1.60E-1	$Z_L C$
B51-480VAC-LF * B61-480VAC-LF * CST-TNK-LF	*	*	9.32E-3	9.50E-2	$\bar{Z}_L C$
B51-480VAC-LF * B6-480VAC-LF * CST-TNK-LF	*	*	9.32E-3	9.50E-2	$\bar{Z}_L C$
B5-480VAC-LF * B61-480VAC-LF * CST-TNK-LF	*	*	9.32E-3	9.50E-2	$\bar{Z}_L C$
B5-480VAC-LF * B6-480VAC-LF * CST-TNK-LF	*	*	9.32E-3	9.50E-2	$Z_L C$
A3-4160VAC-LF * B61-480VAC-LF * CST-TNK-LF	*	*	9.32E-3	9.50E-2	$\bar{Z}_L C$
A3-4160VAC-LF * B6-480VAC-LF * CST-TNK-LF	*	*	9.32E-3	9.50E-2	$Z_L C$
B51-480VAC-LF * A4-4160VAC-LF * CST-TNK-LF	*	*	9.32E-3	9.50E-2	$\bar{Z}_L C$

* Denotes negligible

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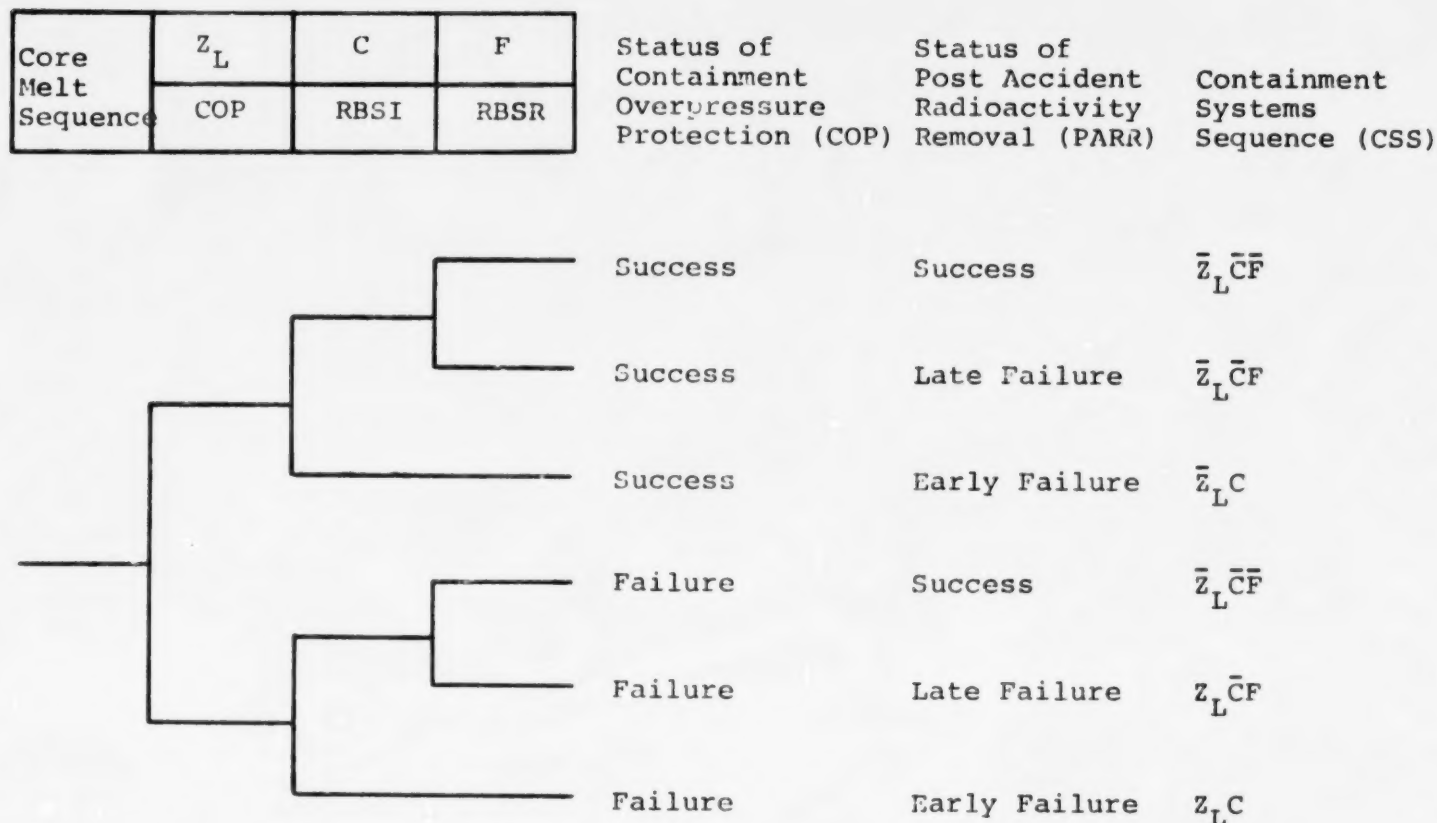
Table 7.5 Dominant Cut Sets in Sequence $S_2(7)$ and $S_3(7)$ (continued)

	Earthquake Level				Containment Tree Branch
	0.1g	0.2g	0.4g	0.6g	
B51-480VAC-LF * B62-480VAC-LF * CST-TNK-LF	*	*	9.32E-3	9.50E-2	$\bar{Z}_L \bar{C}\bar{F}$
B5-480VAC-LF * A4-4160VAC-LF * CST-TNK-LF	*	*	9.32E-3	9.50E-2	$Z_L C$
B5-480VAC-LF * B62-480VAC-LF * CST-TNK-LF	*	*	9.32E-3	9.50E-2	$Z_L \bar{C}\bar{F}$
A3-4160VAC-LF * A4-4160VAC-LF * CST-TNK-LF	*	*	9.32E-3	9.50E-2	$Z_L C$
A3-4160VAC-LF * B62-480VAC-LF * CST-TNK-LF	*	*	9.32E-3	9.50E-2	$Z_L \bar{C}\bar{F}$
BWST-TNK-LF * CST-TNK-LF	*	*	6.73E-3	9.00E-2	$\bar{Z}_L C$
A3-4160VAC-LF * B61-480VAC-LF * EFW2802B-VOC-LF	9.14E-6	7.29E-5	1.09E-2	8.80E-2	$\bar{Z}_L C$
A3-4160VAC-LF * B6-480VAC-LF * EFW2802B-VOC-LF	9.14E-6	7.29E-5	1.09E-2	8.80E-2	$Z_L C$
A3-4160VAC-LF * A4-4160VAC-LF * EFW2902B-VOC-LF	9.14E-6	7.29E-5	1.09E-2	8.80E-2	$Z_L C$
A3-4160VAC-LF * B62-480VAC-LF * EFW2802B-VOC-LF	9.14E-6	7.29E-5	1.09E-2	8.80E-2	$Z_L \bar{C}\bar{F}$
A3-4160VAC-LF * BWST-TNK-LF * EFW2802B-VOC-LF	*	9.27E-6	2.56E-2	8.32E-2	$\bar{Z}_L C$

* Denotes negligible

Table 7.6 Dominant Cut Sets in Sequence $S_2(4)$ and $S_3(4)$

	Earthquake Level				Containment Tree Branch
	0.1g	0.2g	0.4g	0.6g	
B5-480VAC-LF * B6-480VAC-LF	3.58E-1	8.61E-1	9.93E-1	9.99E-1	$Z_L C$
A3-4160VAC-LF * B6-480VAC-LF	3.58E-1	8.61E-1	9.93E-1	9.99E-1	$Z_L C$
B51-480VAC-LF * B6-480VAC-LF	3.58E-1	8.61E-1	9.93E-1	9.99E-1	$\bar{Z}_L C$
B5-480VAC-LF * A4-4160VAC-LF	3.58E-1	9.61E-1	9.93E-1	9.99E-1	$Z_L C$
A3-4160VAC-LF * A4-4160VAC-LF	3.58E-1	8.61E-1	9.93E-1	9.99E-1	$Z_L C$
B51-480VAC-LF * A4-4160VAC-LF	3.28E-1	8.50E-1	9.92E-1	9.99E-1	$\bar{Z}_L C$
A3-4160VAC-LF * B61-480VAC-LF	3.28E-1	8.50E-1	9.92E-1	9.99E-1	$\bar{Z}_L C$
B5-480VAC-LF * B61-480VAC-LF	3.28E-1	8.50E-1	9.92E-1	9.99E-1	$\bar{Z}_L C$
B51-480VAC-LF * B61-480VAC-LF	3.28E-1	8.50E-1	9.92E-1	9.99E-1	$\bar{Z}_L C$
A3-4160VAC-LF * B62-480VAC-LF	3.28E-1	8.50E-1	9.92E-1	9.99E-1	$Z_L \bar{C}\bar{F}$
B5-480VAC-LF * B62-480VAC-LF	3.28E-1	8.50E-1	9.92E-1	9.99E-1	$Z_L \bar{C}\bar{F}$
B51-480VAC-LF * B62-480VAC-LF	3.28E-1	8.50E-1	9.92E-1	9.99E-1	$\bar{Z}_L \bar{C}\bar{F}$



$Z = Y(C+F+H_1)$ all accident sequences except those containing H_2'

$Z = Y(C+F+H_2')$ accident sequences containing H_2'

where $H_1 \equiv 1/3 \text{ HP} + 1/2 \text{ LP} + 1/2 \text{ HTX}$ $H_2' \equiv 1/2 \text{ LP}$

$\therefore$ assume $Z_L = Y(C+F+H_1)$

Figure 7.4 Containment Systems Event Tree - LOCA

Core Melt Sequence	z_T	C
	COP	RBSI

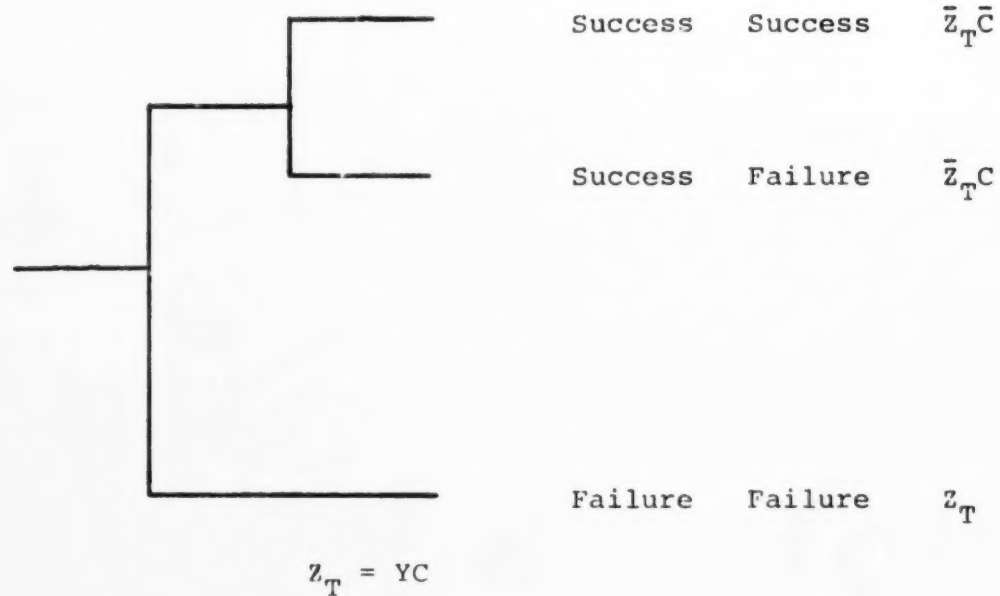


Figure 7.5 Containment Systems Event Tree - Transient

three safety functions Z_L , C and F. Function Z_L denotes the systems available to prevent overpressure of the containment. The overpressure prevention function Z_L (failure) is defined by the Boolean expression

$$Z_L = Y(C + F + H_1)$$

for all accident sequences except those containing H_2' and

$$Z_L = Y(C + F + H_2')$$

for sequences containing H_2' , where

Y = 4 out of 4 containment fan coolers

C = 2 out of 2 RBSI pumps

F = 2 out of 2 RBSR pumps

H_1 = 3 out of 3 HPRS pumps or 2 out of 2 LPRS pumps
or 2 out of 2 heat exchangers

H_2' = 2 out of 2 LPRS pumps

The functions C and F provide the post accident radiation removal function (PARR). The function F denotes failure of both RBSR pumps. The tree defines six outcomes depending on success or failure of the containment safety systems.

Figure 7.5 illustrates the transient containment event tree. The overpressure prevention function Z_T (failure) is defined by the Boolean expression

$$Z_T = YC$$

where Y and C are defined as before.

The function C provides post accident radiation removal (PARR). The tree defines three outcomes depending on success or failure of the containment safety systems.

The mapping from the containment tree sequences to radioactive release categories was based on MARCH code studies performed as part of the Calvert Cliffs RSSMAP study (Reference 17). This plant has a containment size and systems design similar to that at ANO-1.

In order to apportion the accident sequence core damage probabilities to the appropriate branch of the containment event tree, the most general approach is to determine the cut

sets associated with each containment tree branch, and combine these (using Boolean logic) with the accident sequence probabilities.

The assignments of the cutsets in each accident sequence to the containment tree branches are listed in Tables 7.3, 7.4, 7.5, and 7.6. The fractional contribution of each containment branch to the accident sequence probability is computed in Table 7.7.

With these assignments, and the mapping of the containment event tree branches to release categories given in Table 7.8, one can compute the annual probability of release in each category as shown in Table 7.9. Finally, the population dose (in man-rem/year) is estimated using the source term per release category values, taken as 0.3 times the WASH-1400 values, as shown below.

Release Category	Annual Probability	Source Term	Dose (man-rem/yr)
1	7.30E-9	4.7E+5	3.43E-3
2	7.93E-7	4.9E+5	3.89E-1
3	1.04E-5	3.5E+5	3.64E+0
4	5.14E-7	2.1E+5	1.08E-1
5	2.58E-6	1.2E+5	3.10E-1
6	1.91E-5	2.3E+4	4.39E-1
7	5.15E-6	2.4E+3	<u>1.24E-2</u>
Total man-rem/yr			4.9

Thus, for the base case, a total of 4.9 man-rem/year is estimated due to the seismic hazard.

Table 7.7 Partition of Accident Sequence Frequencies to
Containment Branches, ANO-1 Base Case

Sequence	Containment Tree Fractions at Each Earthquake Level				Average* Fraction	Containment Tree Branch
	0.1-0.2	0.2-0.4	0.4-0.6	0.6-up		
$S_2 \bar{M} \bar{L} D_1 \bar{X} D_2$	41.39%	40.21%	40.02%	40%	40.2%	$Z_L C$
	49.13%	49.87%	49.99%	50%	49.9%	$\bar{Z}_L C$
	9.48%	9.92%	9.99%	10%	9.9%	$Z_L \bar{C} \bar{F}$
$S_3 \bar{M} \bar{L} \bar{E} \bar{X} D_2$	41.39%	40.21%	40.02%	40%	40.2%	$Z_L C$
	49.13%	49.87%	49.99%	50%	49.9%	$\bar{Z}_L C$
	9.48%	9.92%	9.99%	10%	9.9%	$Z_L \bar{C} \bar{F}$
$S_2 \bar{M} \bar{L} D_1 X$	34.79%	33.55%	33.35%	33.33%	33.8%	$Z_L C$
	41.30%	41.61%	41.66%	41.67%	41.44%	$\bar{Z}_L C$
	15.94%	16.56%	16.66%	16.67%	16.5%	$Z_L \bar{C} \bar{F}$
	7.97%	8.28%	8.33%	8.33%	8.26%	$\bar{Z}_L \bar{C} \bar{F}$
$S_3 \bar{M} \bar{L} \bar{E} \bar{X}$	34.79%	33.55%	33.35%	33.33%	33.8%	$Z_L C$
	41.30%	41.61%	41.66%	41.67%	41.44%	$\bar{Z}_L C$
	15.94%	16.56%	16.66%	16.67%	16.5%	$Z_L \bar{C} \bar{F}$
	7.97%	8.28%	8.33%	8.33%	8.26%	$\bar{Z}_L \bar{C} \bar{F}$

* Weighted with respect to the accident sequence probability at each level

Table 7.7 Partition of Accident Sequence Frequencies to
Containment Branches, ANO-1 Base Case (continued)

Sequence	Containment Tree Fractions at Each Earthquake Level				Average* Fraction	Containment Tree Branch
	0.1-0.2	0.2-0.4	0.4-0.6	0.6-up		
S_2MLD_1	54.62%	50.72%	42.30%	40.27%	45.32%	$Z_L C$
	22.69%	24.64%	32.88%	38.85%	29.41%	$\bar{Z}_L C$
	22.69%	24.64%	22.38%	16.18%	23.75%	$Z_L \bar{C}\bar{F}$
	0.0%	0.0%	2.44%	4.7%	1.52%	$\bar{Z}_L \bar{C}\bar{F}$
S_3MLE	54.62%	50.72%	42.30%	40.27%	45.32%	$Z_L C$
	22.69%	24.64%	32.88%	38.85%	29.41%	$\bar{Z}_L C$
	22.69%	24.64%	22.38%	16.18%	23.75%	$Z_L \bar{C}\bar{F}$
	0.0%	0.0%	2.44%	4.7%	1.52%	$\bar{Z}_L \bar{C}\bar{F}$
T_2MLPQE	54.62%	50.72%	42.30%	40.27%	49.57%	Z_T
	22.69%	24.64%	32.88%	38.85%	26.16%	$\bar{Z}_T C$
	22.64%	24.64%	24.82%	20.88%	24.27%	$\bar{Z}_T \bar{C}$
T_3MLPQE	54.62%	50.72%	42.30%	40.27%	49.57%	Z_T
	22.69%	24.64%	32.88%	38.85%	26.16%	$\bar{Z}_T C$
	22.64%	24.64%	24.82%	20.88%	24.27%	$\bar{Z}_T \bar{C}$

* Weighted with respect to the accident sequence probability at each level

Table 7.8 Modified Containment Sequences
to Release Category Mapping

Containment Systems Sequence	Special Conditions	Containment Failure Mode with Probability and Release Category				
		α	β	γ, δ_e	δ_l	ϵ
$\bar{ZCF}$		1E-4 1	2E-3 5	1.4E-2 3	1.8E-1 5	2.5E-1 7
$\bar{ZCF}$	EM	1E-4 1	2E-3 5	1.4E-2 3	1.8E-1 4	2.5E-1 6
	LM	1E-4 1	2E-3 4	1.4E-2 2	1.8E-1 3	2.5E-1 6
$\bar{ZC}$		1E-4 1	2E-3 4	1.4E-2 2	1.8E-1 3	2.5E-1 6
ZCF		1E-4 1	2E-3 5	1.4E-2 3	1.8E-1 4	2.5E-1 7
ZCF	EM	1E-4 1	2E-3 5	1.4E-2 3	1.8E-1 4	2.5E-1 7
	LM	1E-4 1	2E-3 4	1.4E-2 2	1.8E-1 3	2.5E-1 7
ZC		1E-4 1	2E-3 4	1.4E-2 2	1.8E-1 3	2.5E-1 6

Table 7.9 Calculation of Annual Release Category Probabilities,
ANO-1, Base Case

Sequence (Probability)	Average Fraction		Release Categories						
			1	2	3	4	5	6	7
$S_2 \bar{M} \bar{L} D_1 \bar{X} D_2$ (2.54E-6)	0.402	$Z_L C$	1.02E-10	1.43E-8	1.83E-7	2.04E-9	*	2.56E-7	*
	0.499	$\bar{Z}_L C$	1.27E-10	1.77E-8	2.27E-7	2.54E-9	*	3.16E-7	*
	0.099	$Z_L \bar{C} \bar{F}$	2.50E-11	*	3.49E-9	4.49E-8	5.02E-10	*	6.25E-8
$S_2 \bar{M} \bar{L} D_1 X$ (7.02E-8)	0.338	$Z_L C$	2.37E-12	3.31E-10	4.27E-9	4.75E-11	*	5.93E-7	*
	0.414	$\bar{Z}_L C$	2.91E-12	4.06E-10	5.23E-9	5.82E-11	*	7.26E-7	*
	0.165	$Z_L \bar{C} \bar{F}$	1.16E-12	*	1.62E-10	2.09E-9	2.32E-11	*	2.88E-7
	0.083	$\bar{Z}_L \bar{C} \bar{F}$	5.82E-13	*	8.14E-11	*	1.06E-9	*	1.46E-9
$S_2 \bar{M} \bar{L} D_1$ (1.54E-6)	0.453	$Z_L C$	6.99E-11	9.78E-9	1.26E-7	1.40E-9	*	1.75E-7	*
	0.294	$\bar{Z}_L C$	4.54E-11	6.35E-9	8.16E-8	9.07E-10	*	1.13E-7	*
	0.238	$Z_L \bar{C} \bar{F}$	3.66E-11	*	5.14E-9	6.59E-8	7.32E-10	*	9.17E-8
	0.015	$\bar{Z}_L \bar{C} \bar{F}$	2.32E-12	*	3.24E-10	*	4.20E-9	*	5.78E-9
$S_3 \bar{h} \bar{L} \bar{E} \bar{X} D_2$ (7.54E-6)	0.402	$Z_L C$	3.04E-10	4.25E-8	5.48E-7	6.08E-9	*	7.64E-7	*
	0.499	$\bar{Z}_L C$	3.77E-10	5.27E-8	6.79E-7	7.56E-9	*	9.44E-7	*
	0.099	$Z_L \bar{C} \bar{F}$	7.48E-11	*	1.04E-8	1.34E-7	1.50E-9	*	1.87E-7

Table 7.9 Calculation of Annual Release Category Probabilities,
ANO-1, Base Case (continued)

Sequence (Probability)	Average Fraction		Release Categories						
			1	2	3	4	5	6	7
S_3 MLEX (1.93E-7)	0.338	$Z_L C$	6.52E-12	9.09E-10	1.17E-8	1.30E-10	*	1.63E-6	*
	0.414	$\bar{Z}_L C$	7.99E-12	1.11E-9	1.44E-8	1.60E-10	*	1.99E-6	*
	0.165	$Z_L \bar{C} \bar{F}$	3.19E-12	*	4.44E-10	5.73E-9	6.38E-11	*	7.92E-6
	0.083	$\bar{Z}_L \bar{C} \bar{F}$	1.60E-12	*	2.24E-10	*	2.91E-9	*	3.99E-9
S_3 MLE (3.03E-6)	0.453	$Z_L C$	1.38E-10	1.93E-8	2.48E-7	2.76E-9	*	3.45E-7	*
	0.294	$\bar{Z}_L C$	8.96E-11	1.25E-8	1.61E-7	1.79E-9	*	2.24E-7	*
	0.238	$Z_L \bar{C} \bar{F}$	7.24E-11	*	1.02E-8	1.30E-7	1.45E-9	*	1.81E-7
	0.015	$\bar{Z}_L \bar{C} \bar{F}$	4.57E-12	*	6.41E-10	*	8.30E-9	*	1.14E-8
T_2 MLPQE (5.08E-5)	0.496	Z_T	2.52E-9	3.52E-7	4.53E-6	5.03E-8	*	6.29E-6	*
	0.261	$\bar{Z}_T C$	1.32E-9	1.85E-7	2.38E-6	2.65E-8	*	3.31E-6	*
	0.243	$\bar{Z}_T \bar{C}$	1.23E-9	*	1.72E-7	*	2.25E-6	*	3.09E-6
T_3 MLPQE (7.26E-6)	0.496	Z_T	3.63E-10	5.08E-8	6.53E-7	7.26E-9	*	9.07E-7	*
	0.261	$\bar{Z}_T C$	1.90E-10	2.67E-8	3.43E-7	3.82E-9	*	4.78E-7	*
	0.243	$\bar{Z}_T \bar{C}$	1.78E-10	*	2.48E-8	*	3.24E-7	*	4.45E-7
TOTAL			7.30E-9	7.93E-7	1.04E-5	5.14E-7	2.58E-6	1.91E-5	5.15E-6

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8.0 POTENTIAL DESIGN CHANGES AND RISK IMPACT

Based on the plant walk-through and the base case evaluation of the risk of seismically-induced core melt at the ANO-1 plant, a number of potential modifications were identified, and the potential reduction in risk resulting from these modifications was evaluated as discussed below.

8.1 Potential Modifications

The BWST is located in the yard (elevation 353 ft). It is predicted to have low seismic resistance, with the failure mode of shell buckling near the base and anchor bolt yielding. The proposed modification to this tank would be to add structural stiffening so as to increase the median ground acceleration level at failure to approximately 4 times the SSE which would significantly reduce its contribution to the probability of core damage.

The second proposed modification would be to anchor the 4160 VAC and the 480 VAC switchgear to the floor, and also, cabinet to cabinet.

Table 8.1 presents the recalculated core melt probabilities for the varying earthquake levels if all of these modifications were implemented. In these calculations, it is assumed that the probabilities of failure of the aforementioned items are reduced to generic levels. In the case of the 4160 VAC and 480 VAC switchgear the generic circuit breaker failure data is used from Attachment C.1. The total core melt probability with these modifications incorporated would be reduced to approximately $8.83\text{E-}6$ per year.

Tables 8.2 through 8.8 map sequence cut sets to containment tree branches. Table 8.9 presents the assignment of proportions of containment tree branches for each seismic level in all dominant sequences.

Table 8.10 shows the mapping of the core damage sequence annual probabilities using the mapping given in Table 7.8. Finally, the population dose (in man-rem/year) is again estimated using the source term per release category values, taken as 0.3 times the WASH-1400 values, as shown below:

<u>Release Category</u>	<u>Annual Probability</u>	<u>Source Term</u>	<u>Dose (man-rem/yr)</u>
1	6.76E-9	4.7E+5	3.18E-3
2	1.17E-7	4.9E+5	5.73E-2
3	1.30E-6	3.5E+5	4.55E-1
4	1.70E-8	2.1E+5	3.57E-3
5	8.44E-8	1.2E+5	1.01E-2
6	2.10E-6	2.3E-4	4.83E-2
7	1.97E-7	2.4E+3	4.73E-4
		Total	6.0E-1

Table 8.1 Accident Sequence Probabilities per Year -
All Modifications Made - ANO-1

Sequence	Earthquake Magnitude				Total
	0.100 - 0.200q	0.200 - 0.400q	0.400 - 0.600q	Above 0.600q	
SLOCA(1)	1.55E-9	6.77E-8	1.02E-7	5.12E-11	1.71E-7
SLOCA(2)	0.0	0.0	4.63E-9	5.60E-14	4.63E-9
SLOCA(3)	4.05E-9	2.46E-7	4.20E-7	1.33E-10	6.70E-7
SLOCA(4)	4.28E-9	2.44E-7	2.55E-7	1.34E-10	5.03E-7
SLOCA(5)	0.0	2.45E-9	3.18E-8	4.88E-13	3.43E-8
SLOCA(7)	0.0	4.12E-8	3.03E-7	7.20E-12	3.44E-7
T2(2)	1.01E-7	1.15E-6	2.62E-6	4.32E-9	3.87E-6
T3(2)	1.12E-7	8.71E-9	0.0	1.97E-9	1.23E-7
SSLOCA(1)	1.77E-8	1.18E-7	1.90E-7	5.04E-10	3.26E-7
SSLOCA(2)	0.0	0.0	8.69E-9	6.16E-13	8.70E-9
SSLOCA(3)	4.78E-8	3.69E-7	7.37E-7	1.38E-9	1.16E-6
SSLOCA(4)	5.45E-8	3.75E-7	4.93E-7	1.57E-9	9.24E-7
SSLOCA(5)	0.0	3.49E-9	6.34E-8	5.76E-12	6.69E-8
SSLOCA(7)	2.64E-9	6.62E-8	5.65E-7	8.00E-11	6.34E-7
TOTAL	3.45E-7	2.69E-6	5.79E-6	1.01E-8	8.83E-6

Table 8.2 Dominant Cut Sets in Sequence $S_2(1)$ and $S_3(1)$

	Earthquake Level				Containment Tree Branch
	0.1g	0.2g	0.4g	0.6g	
SWSV-02-03-CM	1.10E-3	1.10E-3	3.00E-3	3.00E-2	$\bar{Z}_L \bar{C}\bar{F}$
SWSV-40-41-CM	1.10E-3	1.10E-3	3.00E-3	3.00E-2	$\bar{Z}_L \bar{C}\bar{F}$
LPRSV-CM	1.10E-3	1.10E-3	3.00E-3	3.00E-2	$\bar{Z}_L \bar{C}\bar{F}$ (LM)
BATTD07-ZBT-LF * SWS3841-NCC-LF	1.44E-5	5.12E-5	1.86E-3	2.78E-2	$\bar{Z}_L \bar{C}\bar{F}$
BATTD07-ZBT-LF * LPR1406B-VCC-LF	1.44E-5	5.12E-5	1.86E-3	2.78E-2	$\bar{Z}_L \bar{C}\bar{F}$ (LM)
BATTD06-ZBT-LF * SWS3803-VCC-LF	1.44E-5	5.12E-5	1.86E-3	2.78E-2	$\bar{Z}_L \bar{C}\bar{F}$
BATTD06-ZBT-LF * LPR1405A-VCC-LF	1.44E-5	5.12E-5	1.86E-3	2.78E-2	$\bar{Z}_L \bar{C}\bar{F}$ (LM)
BATTD07-ZBT-LF * SWS3802-VCC-LF	1.44E-5	5.12E-5	1.86E-3	2.78E-2	$\bar{Z}_L \bar{C}\bar{F}$
BATTD06-ZBT-LF * SWS3840-NCC-LF	1.44E-5	5.12E-5	1.86E-3	2.78E-2	$\bar{Z}_L \bar{C}\bar{F}$
BATTD06-ZBT-LF * LPI1428A-NOC-LF	*	*	1.08E-3	2.55E-2	$\bar{Z}_L \bar{C}\bar{F}$
BATTD07-ZBT-LF * LPR1415B-VOC-LF	*	*	1.08E-3	2.55E-2	$\bar{Z}_L \bar{C}\bar{F}$ (LM)
BATTD07-ZBT-LF * LPI1429B-NOC-LF	*	*	1.08E-3	2.55E-2	$\bar{Z}_L \bar{C}\bar{F}$

* Denotes negligible

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Table 8.2 Dominant Cut Sets in Sequence $S_2(1)$ and $S_3(1)$ (continued)

	Earthquake Level				Containment Tree Branch
	0.1g	0.2g	0.4g	0.6g	
BATT006-ZBT-L ¹ * LPR1414A-VOC-LF	*	*	1.08E-3	2.55E-2	$\bar{Z}_L \bar{C} \bar{F}$ (LM)
LF-SWS-VCH4B * SWS3802-VCC-LF	1.52E-4	1.52E-4	8.55E-4	1.73E-2	$\bar{Z}_L \bar{C} \bar{F}$
LF-SWS-VCH4A * SWS3840-NCC-LF	1.52E-4	1.52E-4	8.55E-4	1.73E-2	$\bar{Z}_L \bar{C} \bar{F}$
LF-SWS-VCH4B * SWS3841-NCC-LF	1.52E-4	1.52E-4	8.55E-4	1.73E-2	$\bar{Z}_L \bar{C} \bar{F}$
LF-SWS-VCH4B * LPR1406B-VCC-LF	1.52E-4	1.52E-4	8.55E-4	1.73E-2	$\bar{Z}_L \bar{C} \bar{F}$ (LM)
LF-SWS-VCH4A * SWS3803-VCC-LF	1.52E-4	1.52E-4	8.55E-4	1.73E-2	$\bar{Z}_L \bar{C} \bar{F}$
LF-SWS-VCH4A * LPR1405A-VCC-LF	1.52E-4	1.52E-4	8.55E-4	1.73E-2	$\bar{Z}_L \bar{C} \bar{F}$ (LM)
LF-SWS-VCH4B * LPR1415B-VOC-LF	*	1.69E-6	4.95E-4	1.58E-2	$\bar{Z}_L \bar{C} \bar{F}$ (LM)
LF-SWS-VCH4B * LPI1429B-NOC-LF	*	1.69E-6	4.95E-4	1.58E-2	$\bar{Z}_L \bar{C} \bar{F}$
LF-SWS-VCH4A * LPR1414A-VOC-LF	*	1.69E-6	4.95E-4	1.58E-2	$Z_L \bar{C} \bar{F}$ (LM)
LF-SWS-VCH4A * LPI1428A-NOC-LF	*	1.69E-6	4.95E-4	1.58E-2	$\bar{Z}_L \bar{C} \bar{F}$
DG1-GEN-LF * SWS3841-NCC-LF	3.04E-4	3.04E-4	1.05E-3	1.54E-2	$\bar{Z}_L \bar{C} \bar{F}$

* Denotes negligible

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Table 8.2 Dominant Cut Sets in Sequence $S_2(1)$ and $S_3(1)$ (continued)

	Earthquake Level				Containment Tree Branch
	0.1g	0.2g	0.4g	0.6g	
DG1-GEN-LF * SWS3802-VCC-LF	3.04E-4	3.04E-4	1.05E-3	1.54E-2	$\bar{Z}_L \bar{C}\bar{F}$
DG2-GEN-LF * SWS3840-NCC-LF	3.04E-4	3.04E-4	1.05E-3	1.54E-2	$\bar{Z}_L \bar{C}\bar{F}$
DG1-GEN-LF * LPR1406B-VCC-LF	3.04E-4	3.04E-4	1.05E-3	1.54E-2	$\bar{Z}_L \bar{C}\bar{F}$ (LM)
DG2-GEN-LF * SWS3803-VCC-LF	3.04E-4	3.04E-4	1.05E-3	1.54E-2	$\bar{Z}_L \bar{C}\bar{F}$
DG2-GEN-LF * LPR1405A-VCC-LF	3.04E-4	3.04E-4	1.05E-3	1.54E-2	$\bar{Z}_L \bar{C}\bar{F}$ (LM)

* Denotes negligible

Table 8.3 Dominant Cut Sets in Sequence $S_2(2)$ and $S_3(2)$

	Earthquake Level				Containment Tree Branch
	0.1g	0.2g	0.4g	0.6g	
DG1-GEN-LF * SWS3802-VCC-LF * HPIP36CB-PMD-LF	2.64E-6	2.64E-6	1.86E-5	3.53E-4	$\bar{Z}_L \bar{C}\bar{F}$
DG1-GEN-LF * LPR1406B-VCC-LF * HPIP36CB-PMD-LF	2.64E-6	2.64E-6	1.86E-5	3.53E-4	$\bar{Z}_L \bar{C}\bar{F} \text{ (LM)}$
LPI1407A-VCC-LF * B62-480VAC-LF	*	*	6.27E-5	2.40E-3	$\bar{Z}_L \bar{C}\bar{F}$
B62-480VAC-LF * SWSOP4BA-PMD-LF	*	*	5.28E-5	1.48E-3	$Z_L \bar{C}\bar{F}$
BATTD07-ZBT-LF * SWS3810B-VCC-LF * LPR1406B-VCC-LF	*	*	3.54E-5	2.67E-3	$\bar{Z}_L \bar{C}\bar{F} \text{ (LM)}$
BATTD07-ZBT-LF * SWS3810B-VCC-LF * SWS3802-VCC-LF	*	*	3.54E-5	2.67E-3	$\bar{Z}_L \bar{C}\bar{F}$
BATTD07-ZBT-LF * LPR1415B-VOC-LF * SWS3810B-VCC-LF	*	*	2.05E-5	2.45E-3	$\bar{Z}_L \bar{C}\bar{F} \text{ (LM)}$
LPI1407A-VCC-LF * B62-480VAC-LF	*	*	*	2.40E-3	$\bar{Z}_L \bar{C}\bar{F}$
LF-SWS-VCH4B * SWS3802-VCC-LF * SWS3810B-VCC-LF	1.21E-6	1.21E-6	1.86E-5	1.66E-3	$\bar{Z}_L \bar{C}\bar{F}$
LF-SWS-VCH4B * LPR1406B-VCC-LF * SWS3810B-VCC-LF	1.21E-6	1.21E-6	1.86E-5	1.66E-3	$\bar{Z}_L \bar{C}\bar{F} \text{ (LM)}$
LF-SWS-VCH4B * LPR1415B-VOC-LF * SWS3810B-VCC-LF	*	*	9.41E-6	1.52E-3	$\bar{Z}_L \bar{C}\bar{F} \text{ (LM)}$
DG1-GEN-LF * SWS3802-VCC-LF * SWS3810B-VCC-LF	2.43E-6	2.43E-6	1.99E-5	1.48E-3	$\bar{Z}_L \bar{C}\bar{F}$

* Denotes negligible

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Table 8.3 Dominant Cut Sets in Sequence $S_2(2)$ and $S_3(2)$ (continued)

	Earthquake Level				Containment Tree Branch
	0.1g	0.2g	0.4g	0.6g	
DG1-GEN-LF * LPR1406B-VCC-LF * SWS3810B-VCC-LF	2.43E-6	2.43E-6	1.99E-5	1.48E-3	$\bar{Z}_L \bar{C} \bar{F}$ (LM)
DG1-GEN-LF * LPR1415B-VOC-LF * SWS3810B-VCC-LF	*	*	1.15E-5	1.35E-3	$\bar{Z}_L \bar{C} \bar{F}$ (LM)
SWSV-02-03-CM * HPSV-CM	*	*	8.10E-6	9.00E-4	$\bar{Z}_L \bar{C} \bar{F}$
LPRSV-CM * HPSV-CM	*	*	8.10E-6	9.00E-4	$\bar{Z}_L \bar{C} \bar{F}$ (LM)
LPSV-CM * HPSV-CM	*	*	8.10E-6	9.00E-4	$\bar{Z}_L \bar{C} \bar{F}$
SWS3806A-VCC-LF * SWS3802-VCC-LF * SWS3810B-VCC-LF	*	*	6.86E-6	8.85E-4	$\bar{Z}_L \bar{C} \bar{F}$
LPI1407A-VCC-LF * LPR1406B-VCC-LF * SWS3810B-VCC-LF	*	*	6.86E-6	8.85E-4	$\bar{Z}_L \bar{C} \bar{F}$ (LM)
LPI1407A VCC-LF * SWS3802-VCC-LF * SWS3810B-VCC-LF	*	*	6.86E-6	8.85E-4	$\bar{Z}_L \bar{C} \bar{F}$
SWS3806A-VCC-LF * LPR1406B-VCC-LF * SWS3810B-VCC-LF	*	*	6.86E-6	8.85E-4	$\bar{Z}_L \bar{C} \bar{F}$ (LM)
SWS3643A-VCC-LF * SWS3802-VCC-LF * SWS3810B-VCC-LF	*	*	5.05E-6	8.39E-4	$\bar{Z}_L \bar{C} \bar{F}$
SWS3643A-VCC-LF * LPR1406B-VCC-LF * SWS3810B-VCC-LF	*	*	5.05E-6	8.39E-4	$\bar{Z}_L \bar{C} \bar{F}$ (LM)
SWS3806A-VCC-LF * LPR1415B-VOC-LF * SWS3810B-VCC-LF	*	*	3.97E-6	8.11E-4	$\bar{Z}_L \bar{C} \bar{F}$ (LM)

* Denotes negligible

Table 8.3 Dominant Cut Sets in Sequence $S_2(2)$ and $S_3(2)$ (continued)

		Earthquake Level				Containment Tree Branch
		0.1g	0.2g	0.4g	0.6g	
SWS3646A-VOC-LF * SWS3802-VCC-LF * SWS3810B-VCC-LF	*	*		3.97E-6	8.11E-4	$\bar{Z}_L \bar{C} \bar{F}$
LPR1406B-VCC-LF * SWS3645A-VOC-LF * SWS3810B-VCC-LF	*	*		3.97E-6	8.11E-4	$\bar{Z}_L \bar{C} \bar{F}$ (LM)
LPR1406B-VCC-LF * SWS3644A-VOC-LF * SWS3810B-VCC-LF	*	*		3.97E-6	8.11E-4	$\bar{Z}_L \bar{C} \bar{F}$ (LM)
LPI1407A-VCC-LF * LPR1415B-VOC-LF * SWS3810B-VCC-LF	*	*		3.97E-6	8.11E-4	$\bar{Z}_L \bar{C} \bar{F}$ (LM)
LPR1406B-VCC-LF * SWS3646A-VOC-LF * SWS3810B-VCC-LF	*	*		3.97E-6	8.11E-4	$\bar{Z}_L \bar{C} \bar{F}$ (LM)
SWS3644A-VOC-LF * SWS3802-VCC-LF * SWS3810B-VCC-LF	*	*		3.97E-6	8.11E-4	$\bar{Z}_L \bar{C} \bar{F}$
SWS3645A-VOC-LF * SWS3802-VCC-LF * SWS3810B-VCC-LF	*	*		3.97E-6	8.11E-4	$\bar{Z}_L \bar{C} \bar{F}$
SWS3643A-VOC-LF * LPR1415B-VOC-LF * SWS3810B-VCC-LF	*	*	*		7.69E-4	$\bar{Z}_L \bar{C} \bar{F}$ (LM)
BATTD06-ZBT-LF * LPR1414A-VOC-LF * HPSV-CM	*	*		2.91E-6	7.66E-4	$\bar{Z}_L \bar{C} \bar{F}$ (LM)
BATTD07-ZBT-LF * LPR1415B-VOC-LF * HPSV-CM	*	*		2.91E-6	7.66E-4	$\bar{Z}_L \bar{C} \bar{F}$ (LM)
BATTD07-ZBT-LF * SWSV-08-10-CM * LPR1415B-VOC-LF	*	*		3.23E-6	7.66E-4	$\bar{Z}_L \bar{C} \bar{F}$ (LM)
LPR1415B-VOC-LF * SWS3646A-VOC-LF * SWS3810B-VCC-LF	*	*		2.30E-6	7.43E-4	$\bar{Z}_L \bar{C} \bar{F}$ (LM)

* Denotes negligible

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Table 8.3 Dominant Cut Sets in Sequence $S_2(2)$ and $S_3(2)$ (continued)

	Earthquake Level				Containment Tree Branch
	0.1g	0.2g	0.4g	0.6g	
LPRI415B-VOC-LF * SWS3645A-VOC-LF * SWS3810B-VCC-LF	*	*	2.30E-6	7.43E-4	$\bar{Z}_L \bar{C}F$ (LM)
LPRI415B-VOC-LF * SWS3644A-VOC-LF * SWS3810B-VCC-LF	*	*	2.30E-6	7.43E-4	$\bar{Z}_L \bar{C}F$ (LM)
BATTD07-ZBT-LF * LPRI406B-VCC-LF * HPIP36CB-PMD-LF	*	*	1.08E-5	6.40E-4	$\bar{Z}_L \bar{C}F$ (LM)
BATTD07-ZBT-LF * SWS3802-VCC-LF * HPIP36CB-PMD-LF	*	*	1.86E-5	6.40E-4	$\bar{Z}_L \bar{C}F$
BATTD07-ZBT-LF * LPRI415B-VOC-LF * HPIP36CB-PMD-LF	*	*	1.86E-5	5.87E-4	$\bar{Z}_L \bar{C}F$ (LM)

* Denotes negligible

Table 8.4 Dominant Cut Sets in Sequence $S_2(3)$ and $S_3(3)$

	Earthquake Level				Containment Tree Branch
	0.1g	0.2g	0.4g	0.6g	
DG-CM	8.90E-3	8.90E-3	1.30E-2	6.60E-2	$Z_L C$
SWSP-CM	6.50E-3	6.50E-3	6.80E-3	1.40E-2	$Z_L C$
HPSIV-CM	1.10E-3	1.10E-3	3.00E-3	3.00E-2	$\bar{Z}_L C$
SWSV-06-07-CM	1.10E-3	1.10E-3	3.00E-3	3.00E-2	$Z_L C$
SWS3824-VOC-LF	3.60E-5	8.90E-5	1.10E-2	8.80E-2	$Z_L C$
BATTD06-ZBT-LF * DG1-GEN-LF	6.84E-5	6.87E-4	1.04E-2	6.58E-2	$Z_L C$
BATTD07-ZBT-LF * DG2-GEN-LF	6.84E-5	6.87E-4	1.04E-2	6.58E-2	$Z_L C$
BATTD07-ZBT-LF * LF-SWS-VCH4A	3.42E-5	1.21E-4	4.41E-3	5.22E-2	$Z_L C$
BATTD06-ZBT-LF * LF-SWS-VCH4B	3.42E-5	1.21E-4	4.41E-3	5.22E-2	$Z_L C$
LF-SWS-VCH4A * LF-SWS-VCH4B	2.63E-3	2.63E-3	9.58E-3	7.64E-2	$Z_L C$
LF-SWS-VCH4B * DG2-GEN-LF	7.22E-4	7.22E-4	2.48E-3	2.88E-2	$Z_L C$
LF-SWS-VCH4A * DG1-GEN-LF	7.22E-4	7.22E-4	2.48E-3	2.88E-2	$Z_L C$
BATTD06-ZBT-LF * SWS3806A-VCC-LF	1.44E-5	5.12E-5	1.86E-3	2.78E-2	$Z_L C$

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Table 8.4 Dominant Cut Sets in Sequence $S_2(3)$ and $S_3(3)$ (continued)

	Earthquake Level				Containment Tree Branch
	0.1g	0.2g	0.4g	0.6g	
BATTD06-ZBT-LF * LPI1407A-VCC-LF	1.44E-5	5.12E-5	1.86E-3	2.78E-2	$\bar{Z}_L C$
BATTD07-ZBT-LF * LPI1408B-VCC-LF	1.44E-5	5.12E-5	1.86E-3	2.78E-2	$\bar{Z}_L C$
BATTD07-ZBT-LF * SWS3807B-VCC-LF	1.44E-5	5.12E-5	1.86E-3	2.78E-2	$\bar{Z}_L C$
BATTD06-ZBT-LF * SWS3820A-VOO-LF	5.40E-6	1.92E-5	1.37E-3	2.64E-2	$Z_L C$
BATTD06-ZBT-LF * SWS3643A-VOO-LF	5.40E-6	1.92E-5	1.37E-3	2.64E-2	$Z_L C$
BATTD06-ZBT-LF * SWS3645A-VOC-LF	*	*	1.08E-3	2.55E-2	$Z_L C$
BATTD06-ZBT-LF * SWS3644A-VOC-LF	*	*	1.08E-3	2.55E-2	$Z_L C$
BATTD07-ZBT-LF * SWS3641B-VOC-LF	*	*	1.08E-3	2.55E-2	$Z_L C$
BATTD06-ZBT-LF * SWS3646A-VOC-LF	*	*	1.08E-3	2.55E-2	$Z_L C$
LF-SWS-VCH4B * SWSOP4CB-PAC-LF	4.91E-4	4.91E-4	1.57E-3	3.30E-2	$Z_L C$
LF-SWS-VCH4A * SWSOP4BA-PAC-LF	4.91E-4	4.91E-4	1.57E-3	3.30E-2	$Z_L C$

* Denotes negligible

Table 8.5 Dominant Cut Sets in Sequence $S_2(4)$ and $S_3(4)$

	Earthquake Level				Containment Tree Branch
	0.1g	0.2g	0.4g	0.6g	
DG-CM	8.90E-3	8.90E-3	1.30E-2	6.60E-2	$Z_L C$
SWSP-CM	6.50E-3	6.50E-3	6.80E-3	1.40E-2	$Z_L C$
HPISV-CM	1.10E-3	1.10E-3	3.00E-3	3.00E-2	$\bar{Z}_L C$
SWSV-06-07-CM	1.10E-3	1.10E-3	3.00E-3	3.00E-2	$Z_L C$
SWS3824-VOC-LF	3.60E-5	8.90E-5	1.10E-2	8.80E-2	$\bar{Z}_L C$
BATTD06-ZBT-LF * DG1-GEN-LF	6.84E-5	6.87E-4	1.04E-2	6.58E-2	$Z_L C$
BATTD07-ZBT-LF * DG2-GEN-LF	6.84E-5	6.87E-4	1.04E-2	6.58E-2	$Z_L C$
BATTD06-ZBT-LF * LF-SWS-VCH4B	3.42E-5	1.21E-4	4.41E-3	5.22E-2	$Z_L C$
BATTD07-ZBT-LF * LF-SWS-VCH4A	3.42E-5	1.21E-4	4.41E-3	5.22E-2	$Z_L C$
LF-SWS-VCH4A * LF-SWS-VCH4B	2.63E-3	2.63E-3	9.58E-3	7.64E-2	$Z_L C$
HPSV-CM	8.20E-4	8.20E-4	2.70E-3	3.00E-2	$\bar{Z}_L \bar{C} \bar{F}$
LF-SWS-VCH4A * DG1-GEN-LF	7.22E-4	7.22E-4	2.48E-3	2.88E-2	$Z_L C$
LF-SWS-VCH4B * DG2-GEN-LF	7.22E-4	7.22E-4	2.48E-3	2.88E-2	$Z_L C$
BATTD07-ZBT-LF * SWS3810B-VCC-LF	1.44E-5	5.12E-5	1.86E-3	2.78E-2	$\bar{Z}_L \bar{C} \bar{F}$

Table 8.5 Dominant Cut Sets in Sequence $S_2(4)$ and $S_3(4)$ (continued)

	Earthquake Level				Containment Tree Branch
	0.1g	0.2g	0.4g	0.6g	
BATTD07-ZBT-LF * SWS3807B-VCC-LF	1.44E-5	5.12E-5	1.86E-3	2.78E-2	$Z_L C$
BATTD06-ZBT-LF * SWS3806A-VCC-LF	1.44E-5	5.12E-5	1.86E-3	2.78E-2	$Z_L C$
BATTD06-ZBT-LF * LPI1407A-VCC-LF	1.44E-5	5.12E-5	1.86E-3	2.78E-2	$\bar{Z}_L C$
BATTD07-ZBT-LF * LPI1408B-VCC-LF	1.44E-5	5.12E-5	1.86E-3	2.78E-2	$\bar{Z}_L C$
BATTD06-ZBT-LF * CWS3820A-VOO-LF	5.40E-6	1.92E-5	1.37E-3	2.64E-2	$Z_L C$
BATTD06-ZBT-LF * SWS3643A-VOO-LF	5.40E-6	1.92E-5	1.37E-3	2.64E-2	$Z_L C$
BATTD07-ZBT-LF * SWS3641B-VOC-LF	*	*	1.08E-3	2.55E-2	$Z_L C$
BATTD06-ZBT-LF * SWS3645A-VOC-LF	*	*	1.08E-3	2.55E-2	$Z_L C$
BATTD06-ZBT-LF * SWS3646A-VOC-LF	*	*	1.08E-3	2.55E-2	$Z_L C$
BATTD06-ZBT-LF * SWS3644A-VOC-LF	*	*	1.08E-3	2.55E-2	$Z_L C$
LF-SWS-VCH4A * SWSOP4BA-PAC-LF	4.91E-4	4.91E-4	1.08E-3	3.30E-2	$Z_L C$
LF-SWS-VCH4B * SWSOP4CB-PAC-LF	4.91E-4	4.91E-4	1.08E-3	3.30E-2	$Z_L C$

* Denotes negligible

Table 8.6 Dominant Cut Sets in Sequence $S_2(5)$ and $S_3(5)$

	Earthquake Level				Containment Tree Branch
	0.1g	0.2g	0.4g	0.6g	
DG1-GEN-LF * LPIOP34B-PMD-LF * EFWOP7AX-PTD-LF	1.16E-5	1.16E-5	1.98E-5	1.80E-4	$\bar{Z}_L \bar{C}\bar{F}$
DG1-GEN-LF * EFWOP7AX-PTD-LF * SWS3802-VCC-LF	1.06E-5	1.06E-5	3.76E-5	7.53E-4	$\bar{Z}_L \bar{C}\bar{F}$
DG1-GEN-LF * EFWOP7AX-PTD-LF * LPR1406B-VCC-LF	1.06E-5	1.06E-5	3.76E-5	7.53E-4	$\bar{Z}_L \bar{C}\bar{F}$ (LM)
DG1-GEN-LF * EFWOP7AX-PTD-LF * SWS3841-NCC-LF	1.06E-5	1.06E-5	3.76E-5	7.53E-4	$\bar{Z}_L \bar{C}\bar{F}$
DG1-GEN-LF * EFWOP7AX-PTD-LF * SWS3821-VCC-LF	1.06E-5	1.06E-5	3.76E-5	7.53E-4	$\bar{Z}_L \bar{C}\bar{F}$
BATTD07-ZBT-LF * EFWOP7AX-PTD-LF * SWS3802-VCC-LF	*	1.79E-6	6.70E-5	1.36E-3	$\bar{Z}_L \bar{C}\bar{F}$
BATTD07-ZBT-LF * EFWOP7AX-PTD-LF * LPR1406B-VCC-LF	*	1.79E-6	6.70E-5	1.36E-3	$\bar{Z}_L \bar{C}\bar{F}$ (LM)
BATTD07-ZBT-LF * EFWOP7AX-PTD-LF * SWS3841-NCC-LF	*	1.79E-6	6.70E-5	1.36E-3	$\bar{Z}_L \bar{C}\bar{F}$
BATTD07-ZBT-LF * EFWOP7AX-PTD-LF * SWS3821-VCC-LF	*	1.79E-6	6.70E-5	1.36E-3	$\bar{Z}_L \bar{C}\bar{F}$
SWSV-02-03-CM * CST-TNK-LF	*	*	2.82E-5	2.85E-3	$\bar{Z}_L \bar{C}\bar{F}$
LPRSV-CM * CST-TNK-LF	*	*	2.82E-5	2.85E-3	$\bar{Z}_L \bar{C}\bar{F}$ (LM)
SWSV-21-22-CM * CST-TNK-LF	*	*	2.82E-5	2.85E-3	$\bar{Z}_L \bar{C}\bar{F}$
SWSV-40-41-CM * CST-TNK-LF	*	*	2.82E-5	2.85E-3	$\bar{Z}_L \bar{C}\bar{F}$

* Denotes negligible

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Table 8.6 Dominant Cut Sets in Sequence $S_2(5)$ and $S_3(5)$ (continued)

	Earthquake Level				Containment Tree Branch
	0.1g	0.2g	0.4g	0.6g	
BATTD07-ZBT-LF * SWS3841-NCC-LF * EFW00Y2B-VCC-LF	*	*	3.54E-5	2.67E-3	$\bar{Z}_L \bar{C}\bar{F}$
BATTD07-ZBT-LF * SWS3802-VCC-LF * EFW00Y2B-VCC-LF	*	*	3.54E-5	2.67E-3	$\bar{Z}_L \bar{C}\bar{F}$
BATTD07-ZBT-LF * SWS3821-VCC-LF * EFW00Y2B-VCC-LF	*	*	3.54E-5	2.67E-3	$\bar{Z}_L \bar{C}\bar{F}$
BATTD07-ZBT-LF * LPRI406B-VCC-LF * EFW00Y2B-VCC-LF	*	*	3.54E-5	2.67E-3	$\bar{Z}_L \bar{C}\bar{F}$ (LM)
BATTD06-ZBT-LF * LPRI405A-VCC-LF * CST-TNK-LF	*	*	1.75E-5	2.65E-3	$\bar{Z}_L \bar{C}\bar{F}$ (LM)
BATTD07-ZBT-LF * SWS3841-NCC-LF * CST-TNK-LF	*	*	1.75E-5	2.65E-3	$\bar{Z}_L \bar{C}\bar{F}$
BATTD07-ZBT-LF * SWS3821-VCC-LF * CST-TNK-LF	*	*	1.75E-5	2.65E-3	$\bar{Z}_L \bar{C}\bar{F}$
BATTD06-ZBT-LF * SWS3840-NCC-LF * CST-TNK-LF	*	*	1.75E-5	2.65E-3	$\bar{Z}_L \bar{C}\bar{F}$
BATTD06-ZBT-LF * SWS3822-VCC-LF * CST-TNK-LF	*	*	1.75E-5	2.65E-3	$\bar{Z}_L \bar{C}\bar{F}$
BATTD06-ZBT-LF * SWS3803-VCC-LF * CST-TNK-LF	*	*	1.75E-5	2.65E-3	$\bar{Z}_L \bar{C}\bar{F}$
BATTD06-ZBT-LF * SWS3808A-VCC-LF * CST-TNK-LF	*	*	1.75E-5	2.65E-3	$\bar{Z}_L \bar{C}\bar{F}$
BATTD07-ZBT-LF * SWS3802-VCC-LF * CST-TNK-LF	*	*	1.75E-5	2.65E-3	$\bar{Z}_L \bar{C}\bar{F}$
BATTD07-ZBT-LF * LPRI406B-VCC-LF * CST-TNK-LF	*	*	1.75E-5	2.65E-3	$\bar{Z}_L \bar{C}\bar{F}$ (LM)

* Denotes negligible

Table 8.6 Dominant Cut Sets in Sequence $S_2(5)$ and $S_3(5)$ (continued)

	Earthquake Level				Containment Tree Branch
	0.1g	0.2g	0.4g	0.6g	
BATTD07-ZB1-LF * SWS3841-NCC-LF * EFW2802B-VOC-LF	*	*	2.05E-5	2.45E-3	$\bar{Z}_L \bar{C}\bar{F}$
BATTD07-ZB1-LF * SWS3802-VCC-LF * EFW2802B-VOC-LF	*	*	2.05E-5	2.45E-3	$\bar{Z}_L \bar{C}\bar{F}$
BATTD07-ZB1-LF * SWS3821-VCC-LF * EFW2802B-VOC-LF	*	*	2.05E-5	2.45E-3	$\bar{Z}_L \bar{C}\bar{F}$
BATTD07-ZB1-LF * LPR1415B-VOC-LF * EFW00Y2B-VCC-LF	*	*	2.05E-5	2.45E-3	$\bar{Z}_L \bar{C}\bar{F}$ (LM)
BATTD07-ZB1-LF * LPR1406B-VCC-LF * EFW2802B-VOC-LF	*	*	2.05E-5	2.45E-3	$\bar{Z}_L \bar{C}\bar{F}$ (LM)
BATTD07-ZB1-LF * LPI1429B-NOC-LF * EFW00Y2B-VCC-LF	*	*	2.05E-5	2.45E-3	$\bar{Z}_L \bar{C}\bar{F}$
LF-SWS-VCH4B * LPIOP34B-PMD * EFWOP7AX-PTD-LF	5.79E-6	5.79E-6	1.62E-5	2.02E-4	$\bar{Z}_L \bar{C}\bar{F}$
LF-SWS-VCH4B * SWS3841NCC-LF * EFWOP7AX-PTD-LF	5.32E-6	5.32E-6	3.07E-5	8.47E-4	$\bar{Z}_L \bar{C}\bar{F}$
LF-SWS-VCH4B * SWS3821-VCC-LF * EFWOP7AX-PTD-LF	5.32E-6	5.32E-6	3.07E-5	8.47E-4	$\bar{Z}_L \bar{C}\bar{F}$
LF-SWS-VCH4B * SWS3802-VCC-LF * EFWOP7AX-PTD-LF	5.32E-6	5.32E-6	3.07E-5	8.47E-4	$\bar{Z}_L \bar{C}\bar{F}$
LF-SWS-VCH4B * LPR1406B-VCC-LF * EFWOP7AX-PTD-LF	5.32E-6	5.32E-6	3.07E-5	8.47E-4	$\bar{Z}_L \bar{C}\bar{F}$ (LM)

* Denotes negligible

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Table 8.7 Dominant Cut Sets in Sequence $S_2(7)$ and $S_3(7)$

	Earthquake Level				Containment Tree Branch
	0.1g	0.2g	0.4g	0.6g	
BATT-CM	5.20E-4	9.50E-4	3.10E-2	1.60E-1	$Z_L C$
DG-CM * EFWOP7AX-PTD-LF	3.12E-4	3.12E-4	4.68E-4	3.23E-3	$Z_L C$
SWSP-CM * EFWOP7AX-PTD-LF	2.28E-4	2.28E-4	2.45E-4	6.86E-4	$Z_L C$
LF-DC-Y22DC * SWS3824-VOC-LF * LF-DC-Y11DC	*	*	9.89E-4	2.93E-2	$Z_L C$
LF-DC-Y22 * SWS3824-VOC-LF * LF-DC-Y11	*	*	9.84E-4	2.93E-2	$Z_L C$
LF-DC-Y22DC * SWS3824-VOC-LF * LF-DC-Y11	*	*	9.84E-4	2.93E-2	$Z_L C$
LF-DC-Y22 * SWS3824-VOC-LF * LF-DC-Y11DC	*	*	9.84E-4	2.93E-2	$Z_L C$
BATTD07-ZBT-LF * LF-DC-Y22DC * SWS3824-VOC-LF	*	*	3.52E-4	1.57E-2	$Z_L C$
BATTD06-ZBT-LF * SWS3824-VOC-LF * LF-DC-Y11DC	*	*	3.52E-4	1.57E-2	$Z_L C$
BATTD07-ZBT-LF * LF-DC-Y22 * SWS3824-VOC-LF	*	*	3.52E-4	1.57E-2	$Z_L C$
BATTD06-ZBT-LF * SWS3824-VOC-LF * LF-DC-Y11	*	*	3.52E-4	1.57E-2	$Z_L C$
SWS3824-VOC-LF * CST-TNK-LF	*	*	1.03E-4	8.37E-3	$Z_L C$
SWS3824-VOC-LF * EFW2802B-VOC-LF	*	*	1.21E-4	7.74E-3	$Z_L C$
DG-CM * CST-TNK-LF	*	*	1.22E-4	6.28E-3	$Z_L C$

* Denotes negligible

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Table 8.7 Dominant Cut Sets in Sequence $S_2(7)$ and $S_3(7)$ (continued)

	Earthquake Level				Containment Tree Branch
	0.1g	0.2g	0.4g	0.6g	
DG-CM * EFW2802B-VCC-LF	*	*	1.43E-4	5.81E-3	$Z_L C$

* Denotes negligible

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Table 8.8 Dominant Cut Sets in Sequence $T_2(2)$ and $T_3(2)$

	Earthquake Level				Containment Tree Branch
	0.1g	0.2g	0.4g	0.6g	
BATT-CM	5.20E-4	9.50E-4	3.10E-2	1.60E-1	Z_T
DG-CM * EFWOP7AX-PTD-LF	3.12E-4	3.12E-4	4.68E-4	3.23E-3	Z_T
SWSP-CM * EFWOP7AX-PTD-LF	2.28E-4	2.28E-4	2.45E-4	6.86E-4	Z_T
LF-DC-Y22DC * SWS3824-VOC-LF * LF-DC-Y11DC	*	*	9.84E-4	2.93E-2	Z_T
LF-DC-Y22 * SWS3824-VOC-LF * LF-DC-Y11	*	*	9.84E-4	2.93E-2	Z_T
LF-DC-Y22DC * SWS3824-VOC-LF * LF-DC-Y11	*	*	9.84E-4	2.93E-2	Z_T
LF-DC-Y22 * SWS3824-VOC-LF * LF-DC-Y11DC	*	*	9.84E-4	2.93E-2	Z_T
BATTD07-ZBT-LF * LF-DC-Y22DC * SWS3824-VOC-LF	*	*	3.52E-4	1.57E-2	Z_T
BATTD06-ZBT-LF * SWS3824-VOC-LF * LF-DC-Y11DC	*	*	3.52E-4	1.57E-2	Z_T
BATTD07-ZBT-LF * LF-DC-Y22 * SWS3824-VOC-LF	*	*	3.52E-4	1.57E-2	Z_T
BATTD06-ZBT-LF * SWS3824-VOC-LF * LF-DC-Y11	*	*	3.52E-4	1.57E-2	Z_T
SWS3824-VOC-LF * CST-TNK-LF	*	*	1.03E-4	8.37E-3	Z_T
SWS3824-VOC-LF * EFW2802B-VOC-LF	*	*	1.21E-4	7.74E-3	Z_T

* Denotes negligible

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Table 8.8 Dominant Cut Sets in Sequence $T_2(2)$ and $T_3(2)$ (continued)

	Earthquake Level				Containment Tree Branch
	0.1g	0.2g	0.4g	0.6g	
DG-CM * CST-TNK-LF	*	*	1.22E-4	6.28E-3	Z_T
DG-CM * EFW2802B-VOC-LF	*	*	1.43E-4	5.81E-3	Z_T

* Denotes negligible

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Table 8.9 Partition of Accident Sequence Frequencies to Containment Branches, ANO-1 Modified Base Case

Sequence	Containment Tree Fractions at Each Earthquake Level				Average* Fraction	Containment Tree Branch
	0.1-0.2	0.2-0.4	0.4-0.6	0.6-up		
$S_2 \bar{M} \bar{L} \bar{D}_1 H_1 H'_2$	33.33%	33.36%	36.10%	37.78%	34.81%	$\bar{Z}_L \bar{C} \bar{F}$ (LM)
	66.67%	66.64%	63.90%	62.22%	65.19%	$\bar{Z}_L \bar{C} \bar{F}$
$S_3 \bar{M} \bar{L} \bar{E} \bar{X} H_1 H'_2$	33.33%	33.36%	36.10%	37.78%	34.81%	$\bar{Z}_L \bar{C} \bar{F}$ (LM)
	66.67%	66.64%	63.90%	62.22%	65.19%	$\bar{Z}_L \bar{C} \bar{F}$
$S_2 \bar{M} \bar{L} \bar{D}_1 \bar{X} \bar{D}_2 H'_2$	50%	50%	45.24%	43.93%	45.26%	$\bar{Z}_L \bar{C} \bar{F}$ (LM)
	50%	50%	44.19%	52.81%	44.21%	$\bar{Z}_L \bar{C} \bar{F}$
	0%	0%	10.57%	3.26%	10.53%	$Z_L \bar{C} \bar{F}$
$S_3 \bar{M} \bar{L} \bar{E} \bar{X} \bar{D}_2 H'_2$	50%	50%	45.24%	43.93%	45.26%	$\bar{Z}_L \bar{C} \bar{F}$ (LM)
	50%	50%	44.19%	52.81%	44.21%	$\bar{Z}_L \bar{C} \bar{F}$
	0%	0%	10.57%	3.26%	10.53%	$Z_L \bar{C} \bar{F}$
$S_2 \bar{M} \bar{L} \bar{D}_1 \bar{X} \bar{D}_2$	5.13%	5.46%	19.86%	19.78%	14.75%	$\bar{Z}_L C$
	94.87%	94.54%	80.14%	80.22%	85.25%	$Z_L C$
$S_3 \bar{M} \bar{L} \bar{E} \bar{X} \bar{D}_2$	5.13%	5.46%	19.86%	19.78%	14.75%	$\bar{Z}_L C$
	94.87%	94.54%	80.14%	80.22%	85.25%	$Z_L C$

* Weighted with respect to the accident sequence probability at each level

Table 8.9 Partition of Accident Sequence Frequencies to
Containment Branches, ANO-1 Modified Base Case (cont.)

Sequence	Containment Tree Fractions at Each Earthquake Level				Average* Fraction	Containment Tree Branch
	0.1-0.2	0.2-0.4	0.4-0.6	0.6-up		
$S_2 \bar{M} \bar{L} D_1 X$	91.61%	91.51%	78.20%	78.49%	84.51%	$Z_L C$
	4.89%	5.07%	17.34%	16.14%	11.48%	$\bar{Z}_L C$
	3.50%	3.42%	4.46%	5.37%	4.01%	$\bar{Z}_L \bar{C} \bar{F}$
$S_3 \bar{M} \bar{L} E X$	91.61%	91.51%	78.20%	78.49%	84.51%	$Z_L C$
	4.89%	5.07%	17.34%	16.14%	11.48%	$\bar{Z}_L C$
	3.50%	3.42%	4.46%	5.37%	4.01%	$\bar{Z}_L \bar{C} \bar{F}$
$S_2 \bar{M} \bar{L} \bar{D}_1 \bar{P}_1 H_1$	80.36%	79.93%	75.28%	74.36%	75.64%	$\bar{Z}_L \bar{C} \bar{F}$
	19.64%	20.07%	24.72%	25.64%	24.36%	$\bar{Z}_L \bar{C} \bar{F} (LM)$
$S_3 \bar{M} \bar{L} \bar{E} \bar{P}_1 H_1$	80.36%	79.93%	75.28%	74.36%	75.64%	$\bar{Z}_L \bar{C} \bar{F}$
	19.64%	20.07%	24.72%	25.64%	24.36%	$\bar{Z}_L \bar{C} \bar{F} (LM)$
$S_2 \bar{M} \bar{L} D_1$	100%	100%	100%	100%	100%	$Z_L C$
$S_3 \bar{M} \bar{L} E$	100%	100%	100%	100%	100%	$Z_L C$
$T_2 \bar{M} \bar{L} \bar{P} \bar{Q} \bar{E}$	100%	100%	100%	100%	100%	Z_T
$T_3 \bar{M} \bar{L} \bar{P} \bar{Q} \bar{E}$	100%	100%	100%	100%	100%	Z_T

* Weighted with respect to the accident sequence probability at each level

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Table 8.10 Calculation of Annual Release Category Probabilities,
ANO-1, All Modifications Made

Sequence (Probability)	Average Fraction		Release Categories						
			1	2	3	4	5	6	7
$S_2 \bar{M} \bar{L} \bar{D}_1 H_1 H_2'$	0.348	$\bar{Z}_L \bar{C} \bar{F}$ (LM)	4.07E-12	5.69E-10	7.31E-9	8.13E-11	*	1.02E-8	*
(1.71E-7)	0.652	$\bar{Z}_L \bar{C} \bar{F}$	7.61E-12	*	1.07E-9	*	1.39E-8	*	1.90E-8
$S_2 \bar{M} \bar{L} \bar{D}_1 \bar{X} \bar{D}_2 H_2'$	0.453	$\bar{Z}_L \bar{C} \bar{F}$ (LM)	2.09E-13	2.93E-11	3.76E-10	4.18E-12	*	5.25E-10	*
(4.63E-9)	0.442	$\bar{Z}_L \bar{C} \bar{F}$	2.05E-9	*	2.86E-11	*	3.72E-10	*	5.12E-10
	0.105	$Z_L \bar{C} \bar{F}$	4.87E-14	*	6.82E-12	8.77E-11	9.74E-13	*	1.22E-10
$S_2 \bar{M} \bar{L} \bar{D}_1 \bar{X} \bar{D}_2$	0.147	$\bar{Z}_L C$	9.85E-12	1.38E-9	1.77E-8	1.97E-10	*	2.46E-8	*
(6.70E-7)	0.853	$Z_L C$	5.71E-11	8.02E-9	1.03E-7	1.14E-9	*	1.43E-7	*
$S_2 \bar{M} \bar{L} \bar{D}_1 X$	0.845	$Z_L C$	4.26E-11	5.95E-9	7.67E-8	8.52E-10	*	1.07E-7	*
(5.03E-7)	0.115	$\bar{Z}_L C$	5.81E-12	8.10E-10	1.04E-8	1.16E-10	*	1.45E-8	*
	0.040	$\bar{Z}_L \bar{C} \bar{F}$	2.01E-12	*	2.82E-10	*	3.66E-9	*	5.03E-9
$S_2 \bar{M} \bar{L} \bar{D}_1 \bar{P}_1 H_1$	0.756	$\bar{Z}_L \bar{C} \bar{F}$	2.60E-12	*	3.64E-10	*	4.73E-9	*	6.49E-9
(3.43E-8)	0.244	$\bar{Z}_L \bar{C} \bar{F}$ (LM)	8.36E-13	1.17E-10	1.51E-9	1.67E-11	*	2.09E-9	*
$S_2 \bar{M} \bar{L} \bar{D}_1$	1.00	$Z_L C$	3.44E-11	4.82E-9	6.20E-8	6.90E-10	*	8.62E-8	*
(3.44E-7)									

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Table 8.10 Calculation of Annual Release Category Probabilities,
ANO-1, All Modifications Made (continued)

Sequence (Probability)	Average Fraction		Release Categories						
			1	2	3	4	5	6	7
$S_3 \bar{M} \bar{L} \bar{E} H_1 H_2'$	0.348	$\bar{Z}_L \bar{C} \bar{F} (LM)$	1.32E-11	1.85E-9	2.38E-8	2.65E-10	*	3.31E-8	*
(3.26E-7)	0.652	$\bar{Z}_L \bar{C} \bar{F}$	2.48E-11	*	3.47E-9	*	4.51E-8	*	6.20E-8
$S_3 \bar{M} \bar{L} \bar{E} \bar{X} \bar{D}_2 H_2'$	0.453	$\bar{Z}_L \bar{C} \bar{F} (LM)$	3.93E-13	5.50E-11	7.04E-10	7.82E-12	*	9.85E-10	*
(8.70E-9)	0.442	$\bar{Z}_L \bar{C} \bar{F}$	3.83E-9	*	5.37E-11	*	5.98E-10	*	9.58E-10
	0.105	$Z_L \bar{C} \bar{F}$	9.13E-14	*	1.28E-11	1.64E-10	1.83E-12	*	2.28E-10
$S_3 \bar{M} \bar{L} \bar{E} \bar{X} \bar{D}_2$	0.147	$\bar{Z}_L C$	1.71E-11	2.39E-9	3.07E-8	3.41E-10	*	4.27E-8	*
(1.16E-6)	0.853	$Z_L C$	9.89E-11	1.39E-8	1.78E-7	1.98E-9	*	2.47E-7	*
$S_3 \bar{M} \bar{L} \bar{E} X$	0.845	$Z_L C$	7.84E-11	1.10E-8	1.41E-7	1.57E-9	*	1.96E-7	*
(9.24E-7)	0.115	$\bar{Z}_L C$	1.07E-11	1.49E-9	1.92E-8	2.13E-10	*	2.67E-8	*
	0.040	$\bar{Z}_L \bar{C} \bar{F}$	3.71E-12	*	5.19E-10	*	6.74E-9	*	9.27E-9
$S_3 \bar{M} \bar{L} \bar{E} \bar{P}_1 H_1$	0.756	$\bar{Z}_L \bar{C} \bar{F}$	5.05E-12	*	7.06E-10	*	9.17E-9	*	1.26E-8
(6.69E-8)	0.244	$\bar{Z}_L \bar{C} \bar{F} (LM)$	1.62E-12	2.27E-10	2.92E-9	3.25E-11	*	4.07E-9	*
$S_3 \bar{M} \bar{L} \bar{E}$	1.00	$Z_L C$	6.34E-11	8.88E-9	1.14E-7	1.27E-9	*	1.59E-7	*
(6.34E-7)									

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Table 8.10 Calculation of Annual Release Category Probabilities,
ANO-1, All Modifications Made (continued)

Sequence (Probability)	Average Fraction		Release Categories						
			1	2	3	4	5	6	7
T_2 MLPQE	1.00	Z_T	3.87E-10	5.42E-8	6.96E-7	7.76E-9	*	9.70E-7	*
(3.87E-6)									
T_3 MLPQE	1.00	Z_T	1.20E-11	1.68E-9	2.15E-8	2.40E-10	*	3.00E-8	*
(1.23E-7)									
TOTAL			6.76E-9	1.17E-7	1.30E-6	1.70E-8	8.44E-8	2.10E-6	1.97E-7

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Thus, for the case when all modifications are made, the estimated annual population dose due to earthquakes alone is 0.6 man-rem/yr, which is an 88% decrease relative to the base case.

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9.0 SUMMARY AND CONCLUSIONS

Two cases of the seismic risk at ANO-1 have been analyzed, a base-case which includes a number of potentially vulnerable site-specific items, and a case with all site-specific vulnerable components upgraded. In both cases credit was taken for feed and bleed cooling and secondary blowdown.

The base case has the largest probability of core damage ($7.3E-5$ per year) and an estimated exposure of 4.9 man-rem/year. The dominant base case sequence is a transient with LOSP (~ 70%) or with PCS lost (~ 10%) involving failure of the EFWS and HPIS. The dominant contributors to this sequence are failures of different combinations of two electrical busses and either the CST or EFW turbine-driven pump. The 4160 VAC and 480 VAC busses were identified as potential site-specific vulnerabilities at ANO-1. The remaining percentage (20%) of core melt frequency is made up of various LOCA initiated sequences. A small-small LOCA initiator with early failures of HPIS and LPIS is a dominant (10%) contributor to core melt within this group. Dominant cut sets in this sequence are combinations of two electrical bus failures.

In the second case analyzed, the 4160 VAC and 480 VAC busses are assumed to be upgraded so that their seismic vulnerabilities are generic circuit breaker values. This assumes good anchorage and that tipping and sliding are no longer the dominant failure mode. In this case, the frequency of core damage drops to $8.8E-6$ per year, and the exposure drops to 0.6 man-rem/year, nearly an 88% decrease in risk. The two most dominant sequences are again $T_2(2)$ at 44% and $S_3(3)$ at 13%. Four other sequences [$S_3(3)$, $S_2(3)$, $S_3(4)$, and $S_2(4)$] contribute between 10% to 7% each. Blackout type cutsets dominate the transient and $S_3(7)$ sequences; while blackout cut sets, or battery failures paired with either a SWS valve or ventilation system fault dominate the $S_3(3)$, $S_2(3)$, $S_3(4)$, and $S_2(4)$ sequences.

In the base case, the dominant failure mode was different combinations of electrical bus failures with or without another event to fail the EFWS. With modifications these types of cut sets no longer dominated. The modified base case was dominated by common mode failures of either pumps or valves, and different combinations of support system failures such as batteries, ventilation, and service water.

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APPENDIX C.1

Generic Component Seismic Failure

Probabilities for SSE = 0.2 g

PAGE = 1

SSE = 0.20

COMPONENT	MEDIAN FRAGIL	LOG SD FRAGIL	LOG SD RESP	MULT FACTOR	1 SSE	SEISMIC FAILURE 2 SSE	PROBABILITY 3 SSE	4 SSE	5 SSE
CERAMIC INSULATORS	0.25	0.25	0.25	0.50	4.8E-03	2.6E-01	7.0E-01	9.1E-01	9.8E-01
				1.00	2.6E-01	9.1E-01	9.9E-01	1.0E+00	1.0E+00
				1.25	5.0E-01	9.8E-01	1.0E+00	1.0E+00	1.0E+00
				1.50	7.0E-01	9.9E-01	1.0E+00	1.0E+00	1.0E+00
				1.75	8.3E-01	1.0E+00	1.0E+00	1.0E+00	1.0E+00
				2.00	9.1E-01	1.0E+00	1.0E+00	1.0E+00	1.0E+00
				2.25	9.5E-01	1.0E+00	1.0E+00	1.0E+00	1.0E+00
				2.50	9.8E-01	1.0E+00	1.0E+00	1.0E+00	1.0E+00
				2.75	9.9E-01	1.0E+00	1.0E+00	1.0E+00	1.0E+00
				3.00	9.9E-01	1.0E+00	1.0E+00	1.0E+00	1.0E+00
				3.25	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00
				3.50	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00
				3.75	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00
				4.00	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00
				4.25	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00
				4.50	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00
				4.75	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00
				5.00	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00
				5.25	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00
				5.50	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00
				5.75	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00
				6.00	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00
RELAYS	4.00	0.48	0.45	0.50	1.0E-08	2.6E-06	4.1E-05	2.3E-04	7.9E-04
				1.00	2.6E-06	2.3E-04	2.0E-03	7.2E-03	1.8E-02
				1.25	1.3E-05	7.9E-04	5.5E-03	1.8E-02	3.9E-02
				1.50	4.1E-05	2.0E-03	1.2E-02	3.4E-02	6.8E-02
				1.75	1.1E-04	4.0E-03	2.1E-02	5.5E-02	1.0E-01
				2.00	2.3E-04	7.2E-03	3.4E-02	8.2E-02	1.5E-01
				2.25	4.5E-04	1.2E-02	4.9E-02	1.1E-01	1.9E-01
				2.50	7.9E-04	1.8E-02	6.8E-02	1.5E-01	2.4E-01
				2.75	1.3E-03	2.5E-02	8.9E-02	1.8E-01	2.8E-01
				3.00	2.0E-03	3.4E-02	1.1E-01	2.2E-01	3.3E-01
				3.25	2.9E-03	4.4E-02	1.4E-01	2.6E-01	3.8E-01
				3.50	4.0E-03	5.5E-02	1.6E-01	2.9E-01	4.2E-01
				3.75	5.5E-03	6.8E-02	1.9E-01	3.3E-01	4.6E-01
				4.00	7.2E-03	8.2E-02	2.2E-01	3.7E-01	5.0E-01
				4.25	9.3E-03	9.7E-02	2.5E-01	4.0E-01	5.4E-01
				4.50	1.2E-02	1.1E-01	2.8E-01	4.4E-01	5.7E-01
				4.75	1.4E-02	1.3E-01	3.0E-01	4.7E-01	6.0E-01
				5.00	1.8E-02	1.5E-01	3.3E-01	5.0E-01	6.3E-01
				5.25	1E-02	1.6E-01	3.6E-01	5.3E-01	6.6E-01
				5.50	5E-02	1.8E-01	3.8E-01	5.6E-01	6.9E-01
				5.75	9E-02	2.0E-01	4.1E-01	5.8E-01	7.1E-01
				6.00	1.4E-02	2.2E-01	4.4E-01	6.1E-01	7.3E-01

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BEST COPY AVAILABLE

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PAGE = 2

SSE = 0.20

COMPONENT	MEDIAN FRAGIL	LOG SD FRAGIL	LOG SD RESP	MULT FACTOR	1 SSE	SEISMIC FAILURE 2 SSE	PROBABILITY 3 SSE	4 SSE	5 SSE
CIRCUIT BREAKERS	7.63	0.48	0.45	0.50	2.2E-11	1.6E-08	4.4E-07	3.7E-06	1.7E-05
				1.00	1.6E-08	3.7E-06	5.6E-05	3.0E-04	1.0E-03
				1.25	1.0E-07	1.7E-05	2.1E-04	1.0E-03	3.0E-03
				1.50	4.4E-07	5.6E-05	5.8E-04	2.5E-03	6.7E-03
				1.75	1.4E-06	1.4E-04	1.3E-03	5.0E-03	1.3E-02
				2.00	3.7E-06	3.0E-04	2.5E-03	8.8E-03	2.1E-02
				2.25	8.5E-06	5.8E-04	4.2E-03	1.4E-02	3.2E-02
				2.50	1.7E-05	1.0E-03	6.7E-03	2.1E-02	4.5E-02
				2.75	3.2E-05	1.6E-03	1.0E-02	2.9E-02	6.0E-02
				3.00	5.6E-05	2.5E-03	1.4E-02	3.9E-02	7.8E-02
				3.25	9.1E-05	3.6E-03	1.9E-02	5.1E-02	9.7E-02
				3.50	1.4E-04	5.0E-03	2.5E-02	6.4E-02	1.2E-01
				3.75	2.1E-04	6.7E-03	3.2E-02	7.8E-02	1.4E-01
				4.00	3.0E-04	8.8E-03	3.9E-02	9.3E-02	1.6E-01
				4.25	4.3E-04	1.1E-02	4.8E-02	1.1E-01	1.9E-01
				4.50	5.8E-04	1.4E-02	5.7E-02	1.3E-01	2.1E-01
				4.75	7.7E-04	1.7E-02	6.7E-02	1.4E-01	2.4E-01
				5.00	1.0E-03	2.1E-02	7.8E-02	1.6E-01	2.6E-01
				5.25	1.3E-03	2.5E-02	8.9E-02	1.8E-01	2.8E-01
				5.50	1.6E-03	2.9E-02	1.0E-01	2.0E-01	3.1E-01
				5.75	2.0E-03	3.4E-02	1.1E-01	2.2E-01	3.3E-01
				6.00	2.5E-03	3.9E-02	1.3E-01	2.4E-01	3.6E-01
BATTERIES	0.80	0.40	0.35	0.50	4.6E-05	4.6E-03	3.2E-02	9.6E-02	1.9E-01
				1.00	4.6E-03	9.6E-02	2.9E-01	5.0E-01	6.6E-01
				1.25	1.4E-02	1.9E-01	4.5E-01	6.6E-01	8.0E-01
				1.50	3.2E-02	2.9E-01	5.9E-01	7.8E-01	8.8E-01
				1.75	6.0E-02	4.0E-01	7.0E-01	8.5E-01	9.3E-01
				2.00	9.6E-02	5.0E-01	7.8E-01	9.0E-01	9.6E-01
				2.25	1.4E-01	5.9E-01	8.4E-01	9.4E-01	9.7E-01
				2.50	1.9E-01	6.6E-01	8.8E-01	9.6E-01	9.8E-01
				2.75	2.4E-01	7.3E-01	9.1E-01	9.7E-01	9.9E-01
				3.00	2.9E-01	7.8E-01	9.4E-01	9.8E-01	9.9E-01
				3.25	3.5E-01	8.2E-01	9.5E-01	9.9E-01	1.0E+00
				3.50	4.9E-01	8.5E-01	9.7E-01	9.9E-01	1.0E+00
				3.75	4.5E-01	8.8E-01	9.7E-01	9.9E-01	1.0E+00
				4.00	5.0E-01	9.0E-01	9.8E-01	1.0E+00	1.0E+00
				4.25	5.5E-01	9.2E-01	9.9E-01	1.0E+00	1.0E+00
				4.50	5.9E-01	9.4E-01	9.9E-01	1.0E+00	1.0E+00
				4.75	6.3E-01	9.5E-01	9.9E-01	1.0E+00	1.0E+00
				5.00	6.6E-01	9.6E-01	9.9E-01	1.0E+00	1.0E+00
				5.25	7.0E-01	9.7E-01	1.0E+00	1.0E+00	1.0E+00
				5.50	7.3E-01	9.7E-01	1.0E+00	1.0E+00	1.0E+00
				5.75	7.5E-01	9.8E-01	1.0E+00	1.0E+00	1.0E+00
				6.00	7.8E-01	9.8E-01	1.0E+00	1.0E+00	1.0E+00

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BEST COPY AVAILABLE

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PAGE = 3

SSE = 0.20

COMPONENT	MEDIAN FRAGIL	LOG SD FRAGIL	LOG SD RES ²	MULT FACTOR	1 SSE	SEISMIC FAILURE 2 SSE	PROBABILITY 3 SSE	4 SSE	5 SSE
BATTERY RACKS	2.29	0.31	0.35	0.50	1.1E-11	9.2E-08	6.9E-06	9.5E-05	5.7E-04
				1.00	9.2E-08	9.5E-05	2.1E-03	1.2E-02	3.8E-02
				1.25	1.1E-06	5.7E-04	8.5E-03	3.8E-02	9.8E-02
				1.50	6.9E-06	2.1E-03	2.3E-02	8.3E-02	1.8E-01
				1.75	2.9E-05	5.6E-03	4.8E-02	1.5E-01	2.8E-01
				2.00	9.5E-05	1.2E-02	8.3E-02	2.2E-01	3.9E-01
				2.25	2.5E-04	2.3E-02	1.3E-01	3.0E-01	4.8E-01
				2.50	5.7E-04	5.8E-02	1.8E-01	3.9E-01	5.7E-01
				2.75	1.1E-03	8.3E-02	2.4E-01	4.7E-01	6.5E-01
				3.00	2.1E-03	1.1E-01	3.0E-01	5.4E-01	7.2E-01
				3.25	3.5E-03	1.5E-01	3.7E-01	6.1E-01	7.7E-01
				3.50	5.6E-03	1.6E-01	4.3E-01	6.7E-01	8.2E-01
				3.75	8.5E-03	2.2E-01	4.8E-01	7.2E-01	8.5E-01
				4.00	1.2E-02	2.6E-01	5.4E-01	7.6E-01	8.8E-01
				4.25	1.7E-02	3.0E-01	5.9E-01	8.0E-01	9.1E-01
				4.50	2.3E-02	3.4E-01	6.4E-01	8.3E-01	9.3E-01
				4.75	3.0E-02	3.8E-01	6.8E-01	8.6E-01	9.4E-01
				5.00	3.8E-02	4.3E-01	7.2E-01	8.8E-01	9.5E-01
				5.25	4.8E-02	4.7E-01	7.5E-01	9.0E-01	9.6E-01
				5.50	5.8E-02	5.0E-01	7.8E-01	9.2E-01	9.7E-01
				5.75	7.0E-02	5.4E-01	8.1E-01	9.3E-01	9.8E-01
				6.00	8.3E-02	5.8E-01	8.3E-01	9.4E-01	9.8E-01
INVERTERS	2.00	0.26	0.45	0.50	4.1E-09	4.7E-06	1.3E-04	9.8E-04	3.8E-03
				1.00	4.7E-06	9.8E-04	1.0E-02	3.9E-02	9.1E-02
				1.25	3.2E-05	3.8E-03	3.0E-02	9.1E-02	1.8E-01
				1.50	1.3E-04	1.0E-02	6.2E-02	1.6E-01	2.9E-01
				1.75	4.0E-04	2.2E-02	1.1E-01	2.5E-01	4.0E-01
				2.00	9.8E-04	3.9E-02	1.6E-01	3.3E-01	5.0E-01
				2.25	2.1E-03	6.2E-02	2.2E-01	4.2E-01	5.9E-01
				2.50	3.8E-03	9.1E-02	2.9E-01	5.0E-01	6.7E-01
				2.75	6.5E-03	1.3E-01	3.6E-01	5.7E-01	7.3E-01
				3.00	1.0E-02	1.6E-01	4.2E-01	6.4E-01	7.8E-01
				3.25	1.5E-02	2.0E-01	4.8E-01	6.9E-01	8.2E-01
				3.50	2.2E-02	2.5E-01	5.4E-01	7.4E-01	8.6E-01
				3.75	3.0E-02	2.9E-01	5.9E-01	7.8E-01	8.9E-01
				4.00	3.9E-02	3.3E-01	6.4E-01	8.2E-01	9.1E-01
				4.25	5.0E-02	3.8E-01	6.8E-01	8.5E-01	9.3E-01
				4.50	6.2E-02	4.2E-01	7.2E-01	8.7E-01	9.4E-01
				4.75	7.6E-02	4.6E-01	7.5E-01	8.9E-01	9.5E-01
				5.00	9.1E-02	5.0E-01	7.8E-01	9.1E-01	9.6E-01
				5.25	1.1E-01	5.4E-01	8.1E-01	9.2E-01	9.7E-01
				5.50	1.3E-01	5.7E-01	8.3E-01	9.4E-01	9.7E-01
				5.75	1.4E-01	6.1E-01	8.5E-01	9.5E-01	9.8E-01
				6.00	1.6E-01	6.4E-01	8.7E-01	9.5E-01	9.8E-01

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BEST COPY AVAILABLE

PAGE = 4

SSE =0.20

COMPONENT	MEDIAN FRAGIL	LOG SD FRAGIL	LOG SD RESP	MULT FACTOR	1 SSE	SEISMIC FAILURE 2 SSE	PROBABILITY 3 SSE	4 SSE	5 SSE
TRANSFORMERS	8.80	0.28	0.45	0.50	1.5E-17	4.7E-13	9.2E-11	2.7E-09	3.1E-08
				1.00	4.7E-13	2.7E-09	2.0E-07	3.0E-06	2.0E-05
				1.25	9.2E-12	3.1E-08	1.7E-06	2.0E-05	1.2E-04
				1.50	9.2E-11	2.0E-07	8.5E-06	8.5E-05	4.2E-04
				1.75	5.9E-10	8.9E-07	3.0E-05	2.6E-04	1.2E-03
				2.00	2.7E-09	3.0E-06	8.5E-05	6.5E-04	2.6E-03
				2.25	1.0E-08	8.5E-06	2.0E-04	1.4E-03	5.0E-03
				2.50	3.1E-08	2.0E-05	4.2E-04	2.6E-03	8.8E-03
				2.75	8.4E-08	4.4E-05	7.9E-04	4.5E-03	1.4E-02
				3.00	2.0E-07	8.5E-05	1.4E-03	7.1E-03	2.1E-02
				3.25	4.4E-07	1.5E-04	2.2E-03	1.1E-02	3.0E-02
				3.50	8.9E-07	2.6E-04	3.4E-03	1.5E-02	4.1E-02
				3.75	1.7E-06	4.2E-04	5.0E-03	2.1E-02	5.4E-02
				4.00	3.0E-06	6.5E-04	7.1E-03	2.8E-02	6.8E-02
				4.25	5.2E-06	9.6E-04	9.7E-03	3.6E-02	8.5E-02
				4.50	8.5E-06	1.4E-03	1.3E-02	4.6E-02	1.0E-01
				4.75	1.3E-05	1.9E-03	1.7E-02	5.7E-02	1.2E-01
				5.00	2.0E-05	2.6E-03	2.1E-02	6.8E-02	1.4E-01
				5.25	3.0E-05	3.4E-03	2.6E-02	8.1E-02	1.6E-01
				5.50	4.4E-05	4.5E-03	3.2E-02	9.5E-02	1.9E-01
				5.75	6.2E-05	5.7E-03	3.9E-02	1.1E-01	2.1E-01
				6.00	8.5E-05	7.1E-03	4.6E-02	1.3E-01	2.3E-01
MOTOR CONTRL CENTERS-MCC	7.63	0.48	0.45	0.50	2.2E-11	1.6E-08	4.4E-07	3.7E-06	1.7E-05
				1.00	1.6E-08	3.7E-06	5.6E-05	3.0E-04	1.0E-03
				1.25	1.0E-07	1.7E-05	2.1E-04	1.0E-03	3.0E-03
				1.50	4.4E-07	5.6E-05	5.8E-04	2.5E-03	6.7E-03
				1.75	1.4E-06	1.4E-04	1.3E-03	5.0E-03	1.3E-02
				2.00	3.7E-06	3.0E-04	2.5E-03	8.8E-03	2.1E-02
				2.25	8.5E-06	5.8E-04	4.2E-03	1.4E-02	3.2E-02
				2.50	1.7E-05	1.0E-03	6.7E-03	2.1E-02	4.5E-02
				2.75	3.2E-05	1.6E-03	1.0E-02	2.9E-02	6.0E-02
				3.00	5.6E-05	2.5E-03	1.4E-02	3.9E-02	7.8E-02
				3.25	9.1E-05	3.6E-03	1.9E-02	5.1E-02	9.7E-02
				3.50	1.4E-04	5.0E-03	2.5E-02	6.4E-02	1.2E-01
				3.75	2.1E-04	6.7E-03	3.2E-02	7.8E-02	1.4E-01
				4.00	3.0E-04	8.8E-03	3.9E-02	9.3E-02	1.6E-01
				4.25	4.3E-04	1.1E-02	4.8E-02	1.1E-01	1.9E-01
				4.50	5.8E-04	1.4E-02	5.7E-02	1.3E-01	2.1E-01
				4.75	7.7E-04	1.7E-02	6.7E-02	1.4E-01	2.4E-01
				5.00	1.0E-03	2.1E-02	7.8E-02	1.6E-01	2.6E-01
				5.25	1.3E-03	2.5E-02	8.9E-02	1.8E-01	2.8E-01
				5.50	1.6E-03	2.9E-02	1.0E-01	2.0E-01	3.1E-01
				5.75	2.0E-03	3.4E-02	1.1E-01	2.2E-01	3.3E-01
				6.00	2.5E-03	3.9E-02	1.3E-01	2.4E-01	3.6E-01

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BEST COPY AVAILABLE

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PAGE = 5

SSE =0.20

COMPONENT	MEDIAN FRAGIL	LOG SD FRAGIL	LOG SD RESP	MULT FACTOR	1 SSE	SEISMIC FAILURE 2 SSE	PROBABILITY 3 SSE	4 SSE	5 SSE
AUX RELAY CABINETS	7.63	0.48	0.45	0.50	2.2E-11	1.6E-08	4.4E-07	3.7E-06	1.7E-05
				1.00	1.6E-08	3.7E-06	5.6E-05	3.0E-04	1.0E-03
				1.25	1.0E-07	1.7E-05	2.1E-04	1.0E-03	3.0E-03
				1.50	4.4E-07	5.6E-05	5.8E-04	2.5E-03	6.7E-03
				1.75	1.4E-06	1.4E-04	1.3E-03	5.0E-03	1.3E-02
				2.00	3.7E-06	3.0E-04	2.5E-03	8.8E-03	2.1E-02
				2.25	8.5E-06	5.8E-04	4.2E-03	1.4E-02	3.2E-02
				2.50	1.7E-05	1.0E-03	6.7E-03	2.1E-02	4.5E-02
				2.75	3.2E-05	1.6E-03	1.0E-02	2.9E-02	6.0E-02
				3.00	5.6E-05	2.5E-03	1.4E-02	3.9E-02	7.8E-02
				3.25	9.1E-05	3.6E-03	1.9E-02	5.1E-02	9.7E-02
				3.50	1.4E-04	5.0E-03	2.5E-02	6.4E-02	1.2E-01
				3.75	2.1E-04	6.7E-03	3.2E-02	7.8E-02	1.4E-01
				4.00	3.0E-04	8.8E-03	3.9E-02	9.3E-02	1.6E-01
				4.25	4.3E-04	1.1E-02	4.8E-02	1.1E-01	1.9E-01
				4.50	5.8E-04	1.4E-02	5.7E-02	1.3E-01	2.1E-01
				4.75	7.7E-04	1.7E-02	6.7E-02	1.4E-01	2.4E-01
				5.00	1.0E-03	2.1E-02	7.8E-02	1.6E-01	2.6E-01
				5.25	1.3E-03	2.5E-02	8.9E-02	1.8E-01	2.8E-01
				5.50	1.6E-03	2.9E-02	1.0E-01	2.0E-01	3.1E-01
				5.75	2.0E-03	3.4E-02	1.1E-01	2.2E-01	3.3E-01
				6.00	2.5E-03	3.9E-02	1.3E-01	2.4E-01	3.6E-01
SWITCHGEAR	6.43	0.29	0.45	0.50	3.7E-15	4.5E-11	5.2E-09	1.1E-07	9.2E-07
				1.00	4.5E-11	1.1E-07	4.7E-06	5.0E-05	2.5E-04
				1.25	6.6E-10	9.2E-07	3.0E-05	2.5E-04	1.1E-03
				1.50	5.2E-09	4.7E-06	1.2E-04	8.6E-04	3.3E-03
				1.75	2.7E-08	1.7E-05	3.6E-04	2.2E-03	7.5E-03
				2.00	1.1E-07	5.0E-05	8.6E-04	4.7E-03	1.5E-02
				2.25	3.4E-07	1.2E-04	1.8E-03	8.7E-03	2.5E-02
				2.50	9.2E-07	2.5E-04	3.3E-03	1.5E-02	3.9E-02
				2.75	2.2E-06	4.9E-04	5.5E-03	2.3E-02	5.6E-02
				3.00	4.7E-06	8.6E-04	8.7E-03	3.3E-02	7.7E-02
				3.25	9.3E-06	1.4E-03	1.3E-02	4.5E-02	1.0E-01
				3.50	1.7E-05	2.2E-03	1.8E-02	6.0E-02	1.3E-01
				3.75	3.0E-05	3.3E-03	2.5E-02	7.7E-02	1.6E-01
				4.00	5.0E-05	4.7E-03	3.3E-02	9.6E-02	1.9E-01
				4.25	7.8E-05	6.5E-03	4.2E-02	1.2E-01	2.2E-01
				4.50	1.2E-04	8.7E-03	5.3E-02	1.4E-01	2.5E-01
				4.75	1.8E-04	1.1E-02	6.4E-02	1.6E-01	2.9E-01
				5.00	2.5E-04	1.5E-02	7.7E-02	1.9E-01	3.2E-01
				5.25	3.6E-04	1.8E-02	9.1E-02	2.1E-01	3.5E-01
				5.50	4.9E-04	2.3E-02	1.1E-01	2.4E-01	3.9E-01
				5.75	6.5E-04	2.7E-02	1.2E-01	2.7E-01	4.2E-01
				6.00	8.6E-04	3.3E-02	1.4E-01	2.9E-01	4.5E-01

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BEST COPY AVAILABLE

PAGE = 6

SSE =0.20

COMPONENT	MEDIAN FRAGIL	LOG SD FRAGIL	LOG SD RESP	MULT FACTOR	1 SSE	SEISMIC FAILURE 2 SSE	PROBABILITY 3 SSE	4 SSE	5 SSE
CABLE TRAYS	2.23	0.34	0.35	0.50	9.9E-11	3.9E-07	2.0E-05	2.1E-04	1.1E-03
				1.00	3.9E-07	2.1E-04	3.6E-03	1.8E-02	5.0E-02
				1.25	3.7E-06	1.1E-03	1.3E-02	5.0E-02	1.2E-01
				1.50	2.0E-05	3.6E-03	3.1E-02	1.0E-01	2.1E-01
				1.75	7.4E-05	8.8E-03	6.1E-02	1.7E-01	3.1E-01
				2.00	2.1E-04	1.8E-02	1.0E-01	2.5E-01	4.1E-01
				2.25	5.2E-04	3.1E-02	1.5E-01	3.3E-01	5.1E-01
				2.50	1.1E-03	5.0E-02	2.1E-01	4.1E-01	5.9E-01
				2.75	2.1E-03	7.4E-02	2.7E-01	4.9E-01	6.7E-01
				3.00	3.6E-03	1.0E-01	3.3E-01	5.6E-01	7.3E-01
				3.25	5.8E-03	1.3E-01	3.9E-01	6.2E-01	7.8E-01
				3.50	8.8E-03	1.7E-01	4.5E-01	6.8E-01	8.2E-01
				3.75	1.3E-02	2.1E-01	5.1E-01	7.3E-01	8.6E-01
				4.00	1.8E-02	2.5E-01	5.6E-01	7.7E-01	8.8E-01
				4.25	2.4E-02	2.9E-01	6.1E-01	8.1E-01	9.1E-01
				4.50	3.1E-02	3.3E-01	6.5E-01	8.4E-01	9.2E-01
				4.75	4.0E-02	3.7E-01	6.9E-01	8.6E-01	9.4E-01
				5.00	5.0E-02	4.1E-01	7.3E-01	8.8E-01	9.5E-01
				5.25	6.1E-02	4.5E-01	7.6E-01	9.0E-01	9.6E-01
				5.50	7.4E-02	4.9E-01	7.9E-01	9.2E-01	9.7E-01
				5.75	8.7E-02	5.3E-01	8.1E-01	9.3E-01	9.7E-01
				6.00	1.0E-01	5.6E-01	8.4E-01	9.4E-01	9.8E-01
CONTRL PANEL & RACKS	11.50	0.48	0.45	0.50	2.8E-13	3.7E-10	1.5E-08	1.7E-07	9.4E-07
				1.00	3.7E-10	1.7E-07	3.6E-06	2.5E-05	1.0E-04
				1.25	3.0E-09	9.4E-07	1.7E-05	1.0E-04	3.7E-04
				1.50	1.5E-08	3.6E-06	5.4E-05	3.0E-04	9.8E-04
				1.75	5.6E-08	1.0E-05	1.4E-04	6.9E-04	2.1E-03
				2.00	1.7E-07	2.5E-05	3.0E-04	1.4E-03	3.9E-03
				2.25	4.2E-07	5.4E-05	5.7E-04	2.4E-03	6.6E-03
				2.50	9.4E-07	1.0E-04	9.8E-04	3.9E-03	1.0E-02
				2.75	1.9E-06	1.8E-04	1.6E-03	6.0E-03	1.5E-02
				3.00	3.6E-06	3.0E-04	2.4E-03	8.6E-03	2.1E-02
				3.25	6.3E-06	4.6E-04	3.5E-03	1.2E-02	2.7E-02
				3.50	1.0E-05	6.9E-04	4.9E-03	1.6E-02	3.5E-02
				3.75	1.7E-05	9.8E-04	6.6E-03	2.1E-02	4.4E-02
				4.00	2.5E-05	1.4E-03	8.6E-03	2.6E-02	5.4E-02
				4.25	3.8E-05	1.8E-03	1.1E-02	3.2E-02	6.5E-02
				4.50	5.4E-05	2.4E-03	1.4E-02	3.9E-02	7.7E-02
				4.75	7.5E-05	3.1E-03	1.7E-02	4.6E-02	8.9E-02
				5.00	1.0E-04	3.9E-03	2.1E-02	5.4E-02	1.0E-01
				5.25	1.4E-04	4.9E-03	2.5E-02	6.3E-02	1.2E-01
				5.50	1.8E-04	6.0E-03	2.9E-02	7.2E-02	1.3E-01
				5.75	2.3E-04	7.2E-03	3.4E-02	8.2E-02	1.5E-01
				6.00	3.0E-04	8.6E-03	3.9E-02	9.2E-02	1.6E-01

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PAGE = 7

SSE =0.20

COMPONENT	MEDIAN FRAGIL	LOG SD FRAGIL	LOG SD RESP	MULT FACTOR	1 SSE	SEISMIC FAILURE 2 SSE	PROBABILITY 3 SSE	4 SSE	5 SSE
LOCAL INSTRUMENTS	7.68	0.20	0.45	0.50	5.9E-19	6.4E-14	2.3E-11	9.8E-10	1.4E-08
				1.00	6.4E-14	9.8E-10	1.1E-07	2.2E-06	1.7E-05
				1.25	1.8E-12	1.4E-08	1.2E-06	1.7E-05	1.1E-04
				1.50	2.3E-11	1.1E-07	6.7E-06	8.2E-05	4.6E-04
				1.75	1.8E-10	5.7E-07	2.7E-05	2.7E-04	1.3E-03
				2.00	9.8E-10	2.2E-06	8.2E-05	7.2E-04	3.1E-03
				2.25	4.2E-09	6.7E-06	2.1E-04	1.6E-03	6.3E-03
				2.50	1.4E-08	1.7E-05	4.6E-04	3.1E-03	1.1E-02
				2.75	4.3E-08	4.0E-05	9.0E-04	5.6E-03	1.9E-02
				3.00	1.1E-07	8.2E-05	1.6E-03	9.1E-03	2.8E-02
				3.25	2.7E-07	1.5E-04	2.7E-03	1.4E-02	4.0E-02
				3.50	5.7E-07	2.7E-04	4.2E-03	2.0E-02	5.5E-02
				3.75	1.2E-06	4.6E-04	6.3E-03	2.8E-02	7.3E-02
				4.00	2.2E-06	7.2E-04	9.1E-03	3.8E-02	9.3E-02
				4.25	3.9E-06	1.1E-03	1.3E-02	4.9E-02	1.1E-01
				4.50	6.7E-06	1.6E-03	1.7E-02	6.2E-02	1.4E-01
				4.75	1.1E-05	2.3E-03	2.2E-02	7.7E-02	1.6E-01
				5.00	1.7E-05	3.1E-03	2.8E-02	9.3E-02	1.9E-01
				5.25	2.7E-05	4.2E-03	3.5E-02	1.1E-01	2.2E-01
				5.50	4.0E-05	5.6E-03	4.3E-02	1.3E-01	2.5E-01
				5.75	5.8E-05	7.2E-03	5.2E-02	1.5E-01	2.8E-01
				6.00	8.2E-05	9.1E-03	6.2E-02	1.7E-01	3.1E-01
DIESEL GENERATORS	1.00	0.25	0.35	0.50	4.3E-08	9.1E-05	2.6E-03	1.7E-02	5.4E-02
				1.00	9.1E-05	1.7E-02	1.2E-01	3.0E-01	5.0E-01
				1.25	6.3E-04	5.4E-02	2.5E-01	5.0E-01	7.0E-01
				1.50	2.6E-03	1.2E-01	4.0E-01	6.6E-01	8.3E-01
				1.75	7.3E-03	2.0E-01	5.5E-01	7.8E-01	9.0E-01
				2.00	1.7E-02	3.0E-01	6.6E-01	8.6E-01	9.5E-01
				2.25	3.2E-02	4.0E-01	7.6E-01	9.1E-01	9.7E-01
				2.50	5.4E-02	5.0E-01	8.3E-01	9.5E-01	9.8E-01
				2.75	8.2E-02	5.9E-01	8.8E-01	9.7E-01	9.9E-01
				3.00	1.2E-01	6.6E-01	9.1E-01	9.8E-01	9.9E-01
				3.25	1.6E-01	7.3E-01	9.4E-01	9.9E-01	1.0E+00
				3.50	2.0E-01	7.8E-01	9.6E-01	9.9E-01	1.0E+00
				3.75	2.5E-01	8.3E-01	9.7E-01	9.9E-01	1.0E+00
				4.00	3.0E-01	8.6E-01	9.8E-01	1.0E+00	1.0E+00
				4.25	3.5E-01	8.9E-01	9.9E-01	1.0E+00	1.0E+00
				4.50	4.0E-01	9.1E-01	9.9E-01	1.0E+00	1.0E+00
				4.75	4.5E-01	9.3E-01	9.9E-01	1.0E+00	1.0E+00
				5.00	5.0E-01	9.5E-01	9.9E-01	1.0E+00	1.0E+00
				5.25	5.5E-01	9.6E-01	1.0E+00	1.0E+00	1.0E+00
				5.50	5.9E-01	9.7E-01	1.0E+00	1.0E+00	1.0E+00
				5.75	6.3E-01	9.7E-01	1.0E+00	1.0E+00	1.0E+00
				6.00	6.6E-01	9.8E-01	1.0E+00	1.0E+00	1.0E+00

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PAGE = 8

SSE = 0.20

COMPONENT	MEDIAN FRAGIL	LOG SD FRAGIL	LOG SD RESP	MULT FACTOR	SEISMIC FAILURE PROBABILITY				
					1 SSE	2 SSE	3 SSE	4 SSE	5 SSE
HORIZONTAL MOTORS	12.10	0.27	0.35	0.50	1.0E-27	8.4E-21	3.0E-17	6.1E-15	2.8E-13
				1.00	8.4E-21	6.1E-15	5.4E-12	4.0E-10	8.5E-09
				1.25	8.4E-19	2.8E-13	1.6E-10	8.5E-09	1.4E-07
				1.50	3.0E-17	5.4E-12	2.1E-09	8.6E-08	1.2E-06
				1.75	5.5E-16	5.7E-11	1.6E-08	5.3E-07	6.1E-06
				2.00	6.1E-15	4.0E-10	1.6E-08	2.4E-06	2.3E-05
				2.25	4.8E-14	2.1E-09	3.5E-07	8.1E-06	7.1E-05
				2.50	2.8E-13	6.5E-09	1.2E-06	2.3E-05	1.8E-04
				2.75	1.3E-12	2.9E-08	3.3E-06	5.8E-05	4.0E-04
				3.00	5.4E-12	8.6E-08	8.1E-06	1.3E-04	8.0E-04
				3.25	1.9E-11	2.2E-07	1.8E-05	2.5E-04	1.5E-03
				3.50	5.7E-11	5.3E-07	3.7E-05	4.6E-04	2.5E-03
				3.75	1.6E-10	1.2E-06	7.1E-05	8.0E-04	4.0E-03
				4.00	4.0E-10	2.4E-06	1.3E-04	1.3E-03	6.1E-03
				4.25	9.4E-10	4.5E-06	2.1E-04	2.0E-03	9.0E-03
				4.50	2.1E-09	8.1E-06	3.5E-04	3.0E-03	1.3E-02
				4.75	4.3E-09	1.4E-05	5.4E-04	4.4E-03	1.7E-02
				5.00	8.5E-09	2.3E-05	8.0E-04	6.1E-03	2.3E-02
				5.25	1.6E-08	3.7E-05	1.2E-03	8.3E-03	2.9E-02
				5.50	2.9E-08	5.8E-05	1.6E-03	1.1E-02	3.7E-02
				5.75	5.1E-08	8.6E-05	2.3E-03	1.4E-02	4.6E-02
				6.00	8.6E-08	1.3E-04	3.0E-03	1.8E-02	5.6E-02
MOTOR-DRIVEN PUMPS/COMPR	2.80	0.25	0.45	0.50	4.8E-11	1.5E-07	7.2E-06	7.8E-05	4.1E-04
				1.00	1.5E-07	7.8E-05	1.4E-03	7.5E-03	2.3E-02
				1.25	1.3E-06	4.1E-04	5.2E-03	2.3E-02	5.9E-02
				1.50	7.2E-06	1.4E-03	1.4E-02	5.0E-02	1.1E-01
				1.75	2.7E-05	3.5E-03	2.8E-02	8.9E-02	1.8E-01
				2.00	7.8E-05	7.5E-03	5.0E-02	1.4E-01	2.6E-01
				2.25	1.9E-04	1.4E-02	7.8E-02	2.0E-01	3.4E-01
				2.50	4.1E-04	2.3E-02	1.1E-01	2.6E-01	4.1E-01
				2.75	7.8E-04	3.5E-02	1.5E-01	3.2E-01	4.9E-01
				3.00	1.4E-03	5.0E-02	2.0E-01	3.8E-01	5.5E-01
				3.25	2.3E-03	6.8E-02	2.4E-01	4.4E-01	6.1E-01
				3.50	3.5E-03	8.9E-02	2.9E-01	5.0E-01	6.7E-01
				3.75	5.2E-03	1.1E-01	3.4E-01	5.5E-01	7.1E-01
				4.00	7.5E-03	1.4E-01	4.3E-01	6.0E-01	7.6E-01
				4.25	1.0E-02	2.0E-01	4.7E-01	6.5E-01	7.9E-01
				4.50	1.4E-02	2.3E-01	5.1E-01	6.9E-01	8.2E-01
				4.75	1.8E-02	2.6E-01	5.5E-01	7.2E-01	8.5E-01
				5.00	2.3E-02	2.9E-01	5.9E-01	7.6E-01	8.7E-01
				5.25	2.8E-02	3.2E-01	6.3E-01	8.1E-01	9.1E-01
				5.50	3.5E-02	3.5E-01	6.6E-01	8.3E-01	9.2E-01
				5.75	4.2E-02	3.8E-01	6.9E-01	8.5E-01	9.3E-01
				6.00	5.0E-02	4.0E-01	7.0E-01	8.6E-01	9.4E-01

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PAGE = 9

SSE =0.20

COMPONENT	MEDIAN FRAGIL	LOG SD FRAGIL	LOG SD RESP	MULT FACTOR	1 SSE	SEISMIC FAILURE 2 SSE	PROBABILITY 3 SSE	4 SSE	5 SSE
LG VERT M-D CENTRIF PUMP	2.21	0.22	0.45	0.50	3.2E-10	8.1E-07	3.3E-05	3.2E-04	1.5E-03
				1.00	8.1E-07	3.2E-04	4.6E-03	2.1E-02	5.7E-02
				1.25	6.8E-06	1.5E-03	1.5E-02	5.7E-02	1.3E-01
				1.50	3.3E-05	4.6E-03	3.6E-02	1.1E-01	2.2E-01
				1.75	1.2E-04	1.1E-02	6.9E-02	1.8E-01	3.2E-01
				2.00	3.2E-04	2.1E-02	1.1E-01	2.6E-01	4.2E-01
				2.25	7.4E-04	3.6E-02	1.6E-01	3.4E-01	5.1E-01
				2.50	1.5E-03	5.7E-02	2.2E-01	4.2E-01	6.0E-01
				2.75	2.7E-03	8.2E-02	2.8E-01	5.0E-01	6.7E-01
				3.00	4.6E-03	1.1E-01	3.4E-01	5.7E-01	7.3E-01
				3.25	7.3E-03	1.4E-01	4.0E-01	6.3E-01	7.8E-01
				3.50	1.1E-02	1.8E-01	4.6E-01	6.8E-01	8.2E-01
				3.75	1.5E-02	2.2E-01	5.1E-01	7.3E-01	8.5E-01
				4.00	2.1E-02	2.6E-01	5.7E-01	7.7E-01	8.8E-01
				4.25	2.8E-02	3.0E-01	6.1E-01	8.1E-01	9.0E-01
				4.50	3.6E-02	3.4E-01	6.6E-01	8.4E-01	9.2E-01
				4.75	4.6E-02	3.9E-01	6.9E-01	8.6E-01	9.4E-01
				5.00	5.7E-02	4.2E-01	7.3E-01	8.8E-01	9.5E-01
				5.25	6.9E-02	4.6E-01	7.6E-01	9.0E-01	9.6E-01
				5.50	8.2E-02	5.0E-01	7.9E-01	9.2E-01	9.7E-01
				5.75	9.6E-02	5.3E-01	8.1E-01	9.3E-01	9.7E-01
				6.00	1.1E-01	5.7E-01	8.4E-01	9.4E-01	9.8E-01
LARGE MOV	6.50	0.26	0.35	0.50	5.1E-22	7.1E-16	8.7E-13	8.0E-11	2.0E-09
				1.00	7.1E-16	8.0E-11	2.3E-08	7.7E-07	8.8E-06
				1.25	3.9E-14	2.0E-09	3.7E-07	8.8E-06	7.8E-05
				1.50	8.7E-13	2.3E-08	2.9E-06	5.3E-05	3.9E-04
				1.75	1.0E-11	1.6E-07	1.5E-05	2.1E-04	1.3E-03
				2.00	8.0E-11	7.7E-07	5.3E-05	6.5E-04	3.4E-03
				2.25	4.5E-10	2.9E-06	1.6E-04	1.6E-03	7.5E-03
				2.50	2.0E-09	8.0E-06	3.9E-04	3.4E-03	1.4E-02
				2.75	7.4E-09	2.3E-05	8.3E-04	6.5E-03	2.4E-02
				3.00	2.3E-08	5.3E-05	1.6E-03	1.1E-02	3.8E-02
				3.25	6.4E-08	1.1E-04	2.9E-03	1.8E-02	5.6E-02
				3.50	1.6E-07	2.1E-04	4.8E-03	2.7E-02	7.8E-02
				3.75	3.7E-07	3.9E-04	7.5E-03	3.8E-02	1.0E-01
				4.00	7.7E-07	6.5E-04	1.1E-02	5.2E-02	1.3E-01
				4.25	1.5E-06	1.0E-03	1.6E-02	6.9E-02	1.6E-01
				4.50	2.9E-06	1.6E-03	2.2E-02	8.8E-02	2.0E-01
				4.75	5.2E-06	2.4E-03	2.9E-02	1.1E-01	2.4E-01
				5.00	8.8E-06	3.4E-03	3.8E-02	1.3E-01	2.7E-01
				5.25	1.5E-05	4.8E-03	4.8E-02	1.6E-01	3.1E-01
				5.50	2.3E-05	6.5E-03	6.0E-02	1.9E-01	3.5E-01
				5.75	3.6E-05	8.6E-03	7.3E-02	2.1E-01	3.9E-01
				6.00	5.3E-05	1.1E-02	8.9E-02	2.4E-01	4.3E-01

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SSE =0.20

COMPONENT	MEDIAN FRAGIL	LOG SD FRAGIL	LOG SD RESP	MULT FACTOR	1 SSE	SEISMIC FAILURE 2 SSE	PROBABILITY 3 SSE	4 SSE	5 SSE
SMALL MOV SMALL AOV	4.83	0.26	0.35	0.50	3.0E-19	1.4E-13	9.2E-11	5.5E-09	9.9E-08
				1.00	1.4E-13	5.5E-09	8.6E-07	1.9E-05	1.5E-04
				1.25	5.5E-12	9.9E-08	9.7E-06	1.5E-04	9.7E-04
				1.50	9.2E-11	8.6E-07	5.8E-05	7.0E-04	3.7E-03
				1.75	8.7E-10	4.7E-06	2.3E-04	2.3E-03	9.9E-03
				2.00	5.5E-09	1.9E-05	7.0E-04	5.6E-03	2.2E-02
				2.25	2.6E-08	5.0E-05	1.7E-03	1.2E-02	4.0E-02
				2.50	9.9E-08	1.5E-04	3.7E-03	2.2E-02	6.5E-02
				2.75	3.1E-07	3.5E-04	6.9E-03	3.6E-02	9.8E-02
				3.00	8.6E-07	7.0E-04	1.2E-02	5.4E-02	1.4E-01
				3.25	2.1E-06	1.3E-03	1.9E-02	7.8E-02	1.8E-01
				3.50	4.7E-06	2.3E-03	2.8E-02	1.1E-01	2.3E-01
				3.75	9.7E-06	3.7E-03	4.0E-02	1.4E-01	2.8E-01
				4.00	1.9E-05	5.6E-03	5.4E-02	1.7E-01	3.3E-01
				4.25	3.4E-05	8.3E-03	7.1E-02	2.1E-01	3.8E-01
				4.50	5.8E-05	1.2E-02	9.1E-02	2.5E-01	4.4E-01
				4.75	9.6E-05	1.6E-02	1.1E-01	2.9E-01	4.8E-01
				5.00	1.5E-04	2.2E-02	1.4E-01	3.3E-01	5.3E-01
				5.25	2.3E-04	2.8E-02	1.6E-01	3.7E-01	5.8E-01
				5.50	3.5E-04	3.6E-02	1.9E-01	4.2E-01	6.2E-01
				5.75	5.0E-04	4.4E-02	2.2E-01	4.6E-01	6.6E-01
				6.00	7.0E-04	5.4E-02	2.5E-01	4.9E-01	6.9E-01
LG PNEUM/HYD VALVE	6.50	0.26	0.35	0.50	5.1E-22	7.1E-16	8.7E-13	8.0E-11	2.0E-09
				1.00	7.1E-16	8.0E-11	2.3E-08	7.7E-07	8.8E-05
				1.25	3.9E-14	2.0E-09	3.7E-07	8.8E-06	7.8E-05
				1.50	8.7E-13	2.3E-08	2.9E-06	5.3E-05	3.9E-04
				1.75	1.0E-11	1.6E-07	1.5E-05	2.1E-04	1.3E-03
				2.00	8.0E-11	7.7E-07	5.3E-05	6.5E-04	3.4E-03
				2.25	4.5E-10	2.9E-06	1.6E-04	1.6E-03	7.5E-03
				2.50	2.0E-09	8.8E-06	3.9E-04	3.4E-03	1.4E-02
				2.75	7.4E-09	2.3E-05	8.3E-04	6.5E-03	2.4E-02
				3.00	2.3E-08	5.3E-05	1.6E-03	1.1E-02	3.8E-02
				3.25	6.4E-08	1.1E-04	2.9E-03	1.8E-02	5.6E-02
				3.50	1.6E-07	2.1E-04	4.8E-03	2.7E-02	7.8E-02
				3.75	3.7E-07	3.9E-04	7.5E-03	3.8E-02	1.0E-01
				4.00	7.7E-07	6.5E-04	1.1E-02	5.2E-02	1.3E-01
				4.25	1.5E-06	1.0E-03	1.6E-02	6.9E-02	1.6E-01
				4.50	2.9E-06	1.6E-03	2.2E-02	8.6E-02	2.0E-01
				4.75	5.2E-06	2.4E-03	2.9E-02	1.1E-01	2.4E-01
				5.00	8.8E-06	3.4E-03	3.8E-02	1.3E-01	2.7E-01
				5.25	1.5E-05	4.8E-03	4.8E-02	1.6E-01	3.1E-01
				5.50	2.3E-05	6.5E-03	6.0E-02	1.9E-01	3.5E-01
				5.75	3.6E-05	8.6E-03	7.3E-02	2.1E-01	3.9E-01
				6.00	5.3E-05	1.1E-02	8.5E-02	2.4E-01	4.3E-01

BEST COPY AVAILABLE

C-140

PAGE =11

SSE =0.20

COMPONENT	MEDIAN FRAGIL	LOG SD FRAGIL	LOG SD RESP	MULT FACTOR	1 SSE	SEISMIC FAILURE 2 SSE	PROBABILITY 3 SSE	4 SSE	5 SSE
LARGE VALVE MAN,CHK,REL	8.90	0.20	0.35	0.50	4.2E-29	2.4E-21	2.1E-17	7.0E-15	4.6E-13
				1.00	2.4E-21	7.0E-15	1.1E-11	1.1E-09	2.9E-08
				1.25	3.9E-19	4.6E-13	4.2E-10	2.9E-08	5.6E-07
				1.50	2.1E-17	1.1E-11	6.6E-09	3.3E-07	5.0E-06
				1.75	5.0E-16	1.4E-10	5.7E-08	2.2E-06	2.7E-05
				2.00	7.0E-15	1.1E-09	3.3E-07	1.0E-05	1.1E-04
				2.25	6.6E-14	6.6E-09	1.4E-06	3.7E-05	3.2E-04
				2.50	4.6E-13	2.9E-08	5.0E-06	1.1E-04	8.2E-04
				2.75	2.5E-12	1.1E-07	1.5E-05	2.6E-04	1.8E-03
				3.00	1.1E-11	3.3E-07	3.7E-05	5.7E-04	3.5E-03
				3.25	4.2E-11	9.1E-07	8.3E-05	1.1E-03	6.2E-03
				3.50	1.4E-10	2.2E-06	1.7E-04	2.1E-03	1.0E-02
				3.75	4.2E-10	5.0E-06	3.2E-04	3.5E-03	1.6E-02
				4.00	1.1E-09	1.0E-05	5.7E-04	5.6E-03	2.4E-02
				4.25	2.8E-09	2.0E-05	9.7E-04	8.5E-03	3.3E-02
				4.50	6.6E-09	3.7E-05	1.5E-03	1.2E-02	4.5E-02
				4.75	1.4E-08	6.4E-05	2.4E-03	1.7E-02	6.0E-02
				5.00	2.9E-08	1.1E-04	3.5E-03	2.4E-02	7.6E-02
				5.25	5.7E-08	1.7E-04	5.0E-03	3.1E-02	9.5E-02
				5.50	1.1E-07	2.6E-04	6.9E-03	4.0E-02	1.2E-01
				5.75	1.9E-07	3.9E-04	9.4E-03	5.1E-02	1.4E-01
				6.00	3.3E-07	5.7E-04	1.2E-02	6.3E-02	1.6E-01
MISC SMALL VALVES	12.50	0.33	0.35	0.50	5.2E-24	4.1E-18	4.5E-15	4.2E-13	1.1E-11
				1.00	4.1E-18	4.2E-13	1.4E-10	5.5E-09	7.6E-08
				1.25	2.1E-16	1.1E-11	2.5E-09	7.6E-08	8.5E-07
				1.50	4.5E-15	1.4E-10	2.3E-08	5.5E-07	5.2E-06
				1.75	5.3E-14	1.0E-09	1.3E-07	2.7E-06	2.2E-05
				2.00	4.2E-13	5.5E-09	5.5E-07	9.6E-06	7.0E-05
				2.25	2.4E-12	2.3E-08	1.9E-06	2.8E-05	1.8E-04
				2.50	1.1E-11	7.6E-08	5.2E-06	7.0E-05	4.1E-04
				2.75	4.2E-11	2.2E-07	1.3E-05	1.5E-04	8.2E-04
				3.00	1.4E-10	5.5E-07	2.8E-05	3.0E-04	1.5E-03
				3.25	4.0E-10	1.3E-06	5.6E-05	5.5E-04	2.6E-03
				3.50	1.0E-09	2.7E-06	1.0E-04	9.3E-04	4.1E-03
				3.75	2.5E-09	5.2E-06	1.8E-04	1.5E-03	6.2E-03
				4.00	5.5E-09	9.6E-06	3.0E-04	2.3E-03	8.9E-03
				4.25	1.1E-08	1.7E-05	4.8E-04	3.4E-03	1.2E-02
				4.50	2.3E-08	2.8E-05	7.2E-04	4.8E-03	1.7E-02
				4.75	4.2E-08	4.5E-05	1.1E-03	6.7E-03	2.2E-02
				5.00	7.6E-08	7.0E-05	1.5E-03	8.9E-03	2.8E-02
				5.25	1.3E-07	1.0E-04	2.1E-03	1.2E-02	3.6E-02
				5.50	2.2E-07	1.5E-04	2.8E-03	1.5E-02	4.4E-02
				5.75	3.5E-07	2.2E-04	3.7E-03	1.9E-02	5.3E-02
				6.00	5.5E-07	3.0E-04	4.8E-03	2.3E-02	6.4E-02

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BEST COPY AVAILABLE

PAGE =12

SSE =0.20

COMPONENT	MEDIAN FRAGIL	LOG SD FRAGIL	LOG SD RESP	MULT FACTOR	1 SSE	SEISMIC FAILURE 2 SSE	PROBABILITY 3 SSE	4 SSE	5 SSE
LG HORIZONTAL VESSELS	3.00	0.30	0.35	0.50	8.0E-14	2.1E-09	2.9E-07	6.2E-06	5.1E-05
				1.00	2.1E-09	6.2E-06	2.4E-04	2.1E-03	8.6E-03
				1.25	3.5E-08	5.1E-05	1.3E-03	8.6E-03	2.9E-02
				1.50	2.9E-07	2.4E-04	4.5E-03	2.3E-02	6.6E-02
				1.75	1.6E-06	8.0E-04	1.1E-02	4.9E-02	1.2E-01
				2.00	6.2E-06	2.1E-03	2.3E-02	8.6E-02	1.9E-01
				2.25	1.9E-05	4.5E-03	4.2E-02	1.3E-01	2.7E-01
				2.50	5.1E-05	8.6E-03	6.6E-02	1.9E-01	3.5E-01
				2.75	1.2E-04	1.5E-02	9.7E-02	2.5E-01	4.3E-01
				3.00	2.4E-04	2.3E-02	1.3E-01	3.1E-01	5.0E-01
				3.25	4.5E-04	3.5E-02	1.8E-01	3.8E-01	5.7E-01
				3.50	8.0E-04	4.9E-02	2.2E-01	4.4E-01	6.3E-01
				3.75	1.3E-03	6.6E-02	2.7E-01	5.0E-01	6.9E-01
				4.00	2.1E-03	8.6E-02	3.1E-01	5.6E-01	7.3E-01
				4.25	3.1E-03	1.1E-01	3.6E-01	6.1E-01	7.8E-01
				4.50	4.5E-03	1.3E-01	4.1E-01	6.5E-01	8.1E-01
				4.75	6.3E-03	1.6E-01	4.6E-01	7.0E-01	8.4E-01
				5.00	8.6E-03	1.9E-01	5.0E-01	7.3E-01	8.7E-01
				5.25	1.1E-02	2.2E-01	5.4E-01	7.7E-01	8.9E-01
				5.50	1.5E-02	2.5E-01	5.8E-01	8.0E-01	9.1E-01
				5.75	1.9E-02	2.8E-01	6.2E-01	8.2E-01	9.2E-01
				6.00	2.3E-02	3.1E-01	6.5E-01	8.5E-01	9.3E-01
SMALL TO MED HX & VESSELS	1.84	0.25	0.35	0.50	6.4E-12	1.2E-07	1.2E-05	1.9E-04	1.2E-03
				1.00	1.2E-07	1.9E-04	4.6E-03	2.6E-02	7.8E-02
				1.25	1.7E-06	1.2E-03	1.8E-02	7.8E-02	1.8E-01
				1.50	1.2E-05	4.6E-03	4.8E-02	1.6E-01	3.2E-01
				1.75	5.7E-05	1.2E-02	9.6E-02	2.6E-01	4.5E-01
				2.00	1.9E-04	2.6E-02	1.6E-01	3.7E-01	5.8E-01
				2.25	5.3E-04	4.8E-02	2.4E-01	4.8E-01	6.8E-01
				2.50	1.2E-03	7.8E-02	3.2E-01	5.8E-01	7.6E-01
				2.75	2.5E-03	1.2E-01	4.0E-01	5.6E-01	8.2E-01
				3.00	4.6E-03	1.6E-01	4.8E-01	7.3E-01	8.7E-01
				3.25	7.8E-03	2.1E-01	5.5E-01	7.9E-01	9.1E-01
				3.50	1.2E-02	2.6E-01	6.2E-01	8.4E-01	9.3E-01
				3.75	1.8E-02	3.2E-01	6.8E-01	8.7E-01	9.5E-01
				4.00	2.6E-02	3.7E-01	7.3E-01	9.0E-01	9.6E-01
				4.25	3.6E-02	4.3E-01	7.8E-01	9.2E-01	9.7E-01
				4.50	4.8E-02	4.8E-01	8.1E-01	9.4E-01	9.8E-01
				4.75	6.2E-02	5.3E-01	8.5E-01	9.5E-01	9.9E-01
				5.00	7.8E-02	5.8E-01	8.7E-01	9.6E-01	9.9E-01
				5.25	9.6E-02	6.2E-01	8.9E-01	9.7E-01	9.9E-01
				5.50	1.2E-01	6.6E-01	9.1E-01	9.8E-01	9.9E-01
				5.75	1.4E-01	7.0E-01	9.3E-01	9.8E-01	1.0E+00
				6.00	1.6E-01	7.3E-01	9.4E-01	9.9E-01	1.0E+00

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BEST COPY AVAILABLE

C-142

PAGE =13

SSE =0.20

COMPONENT	MEDIAN FRAGIL	LOG SD FRAGIL	LOG SD RESP	MULT FACTOR	1 SSE	SEISMIC FAILURE 2 SSE	PROBABILITY 3 SSE	4 SSE	5 SSE
LG VERT STOR VSL-FORMD HD	1.46	0.20	0.35	0.50	1.5E-11	4.1E-07	4.3E-05	6.6E-04	3.9E-03
				1.00	4.1E-07	6.6E-04	1.4E-02	6.6E-02	1.7E-01
				1.25	6.0E-06	3.9E-03	4.9E-02	1.7E-01	3.5E-01
				1.50	4.3E-05	1.4E-02	1.2E-01	3.1E-01	5.3E-01
				1.75	2.0E-04	3.4E-02	2.1E-01	4.6E-01	6.7E-01
				2.00	6.6E-04	6.8E-02	3.1E-01	5.9E-01	7.8E-01
				2.25	1.8E-03	1.2E-01	4.2E-01	7.0E-01	8.6E-01
				2.50	3.9E-03	1.7E-01	5.3E-01	7.8E-01	9.1E-01
				2.75	7.7E-03	2.4E-01	6.2E-01	8.5E-01	9.4E-01
				3.00	1.4E-02	3.1E-01	7.0E-01	8.9E-01	9.6E-01
				3.25	2.2E-02	3.9E-01	7.6E-01	9.2E-01	9.8E-01
				3.50	3.4E-02	4.6E-01	8.2E-01	9.5E-01	9.8E-01
				3.75	4.9E-02	5.3E-01	8.6E-01	9.6E-01	9.9E-01
				4.00	6.8E-02	5.9E-01	8.9E-01	9.7E-01	9.9E-01
				4.25	9.0E-02	6.5E-01	9.2E-01	9.8E-01	1.0E+00
				4.50	1.2E-01	7.0E-01	9.4E-01	9.9E-01	1.0E+00
				4.75	1.4E-01	7.4E-01	9.5E-01	9.9E-01	1.0E+00
				5.00	1.7E-01	7.8E-01	9.6E-01	9.9E-01	1.0E+00
				5.25	2.1E-01	8.2E-01	9.7E-01	1.0E+00	1.0E+00
				5.50	2.4E-01	8.5E-01	9.8E-01	1.0E+00	1.0E+00
				5.75	2.8E-01	8.7E-01	9.8E-01	1.0E+00	1.0E+00
				6.00	3.1E-01	8.9E-01	9.9E-01	1.0E+00	1.0E+00
LG VERT FLAT BOTTOM TANK	0.45	0.25	0.45	0.50	1.7E-03	5.8E-02	2.2E-01	4.1E-01	5.8E-01
				1.00	5.8E-02	4.1E-01	7.1E-01	8.7E-01	9.4E-01
				1.25	1.3E-01	5.8E-01	8.4E-01	9.4E-01	9.8E-01
				1.50	2.2E-01	7.1E-01	9.1E-01	9.7E-01	9.9E-01
				1.75	3.1E-01	8.0E-01	9.5E-01	9.9E-01	1.0E+00
				2.00	4.1E-01	8.7E-01	9.7E-01	9.9E-01	1.0E+00
				2.25	5.0E-01	9.1E-01	9.8E-01	1.0E+00	1.0E+00
				2.50	5.8E-01	9.4E-01	9.9E-01	1.0E+00	1.0E+00
				2.75	6.5E-01	9.6E-01	9.9E-01	1.0E+00	1.0E+00
				3.00	7.1E-01	9.7E-01	1.0E+00	1.0E+00	1.0E+00
				3.25	7.6E-01	9.8E-01	1.0E+00	1.0E+00	1.0E+00
				3.50	8.0E-01	9.9E-01	1.0E+00	1.0E+00	1.0E+00
				3.75	8.4E-01	9.9E-01	1.0E+00	1.0E+00	1.0E+00
				4.00	8.7E-01	9.9E-01	1.0E+00	1.0E+00	1.0E+00
				4.25	8.9E-01	1.0E+00	1.0E+00	1.0E+00	1.0E+00
				4.50	9.1E-01	1.0E+00	1.0E+00	1.0E+00	1.0E+00
				4.75	9.3E-01	1.0E+00	1.0E+00	1.0E+00	1.0E+00
				5.00	9.4E-01	1.0E+00	1.0E+00	1.0E+00	1.0E+00
				5.25	9.5E-01	1.0E+00	1.0E+00	1.0E+00	1.0E+00
				5.50	9.6E-01	1.0E+00	1.0E+00	1.0E+00	1.0E+00
				5.75	9.7E-01	1.0E+00	1.0E+00	1.0E+00	1.0E+00
				6.00	9.7E-01	1.0E+00	1.0E+00	1.0E+00	1.0E+00

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BEST COPY AVAILABLE

SSE =0.20

COMPONENT	MEDIAN FRAGIL	LOG SD FRAGIL	LOG SD RESP	MULT FACTOR	1 SSE	SEISMIC FAILURE 2 SSE	PROBABILITY 3 SSE	4 SSE	5 SSE
COND STORAGE TANK	0.81	0.28	0.45	0.50	4.0E-05	4.2E-03	3.0E-02	9.2E-02	1.8E-01
				1.00	4.2E-03	9.2E-02	2.3E-01	4.9E-01	6.5E-01
				1.25	1.3E-02	1.8E-01	4.4E-01	6.5E-01	7.9E-01
				1.50	3.0E-02	2.9E-01	5.8E-01	7.7E-01	8.8E-01
				1.75	5.7E-02	3.9E-01	6.9E-01	8.5E-01	9.3E-01
				2.00	9.2E-02	4.9E-01	7.7E-01	9.0E-01	9.6E-01
				2.25	1.3E-01	5.8E-01	8.3E-01	9.3E-01	9.7E-01
				2.50	1.8E-01	6.5E-01	8.8E-01	9.6E-01	9.8E-01
				2.75	2.3E-01	7.2E-01	9.1E-01	9.7E-01	9.9E-01
				3.00	2.9E-01	7.7E-01	9.3E-01	9.8E-01	9.9E-01
				3.25	3.4E-01	8.1E-01	9.5E-01	9.9E-01	1.0E+00
				3.50	3.9E-01	8.5E-01	9.6E-01	9.9E-01	1.0E+00
				3.75	4.4E-01	8.8E-01	9.7E-01	9.9E-01	1.0E+00
				4.00	4.9E-01	9.0E-01	9.8E-01	1.0E+00	1.0E+00
				4.25	5.4E-01	9.2E-01	9.8E-01	1.0E+00	1.0E+00
				4.50	5.8E-01	9.3E-01	9.9E-01	1.0E+00	1.0E+00
				4.75	6.2E-01	9.5E-01	9.9E-01	1.0E+00	1.0E+00
				5.00	6.5E-01	9.6E-01	9.9E-01	1.0E+00	1.0E+00
				5.25	6.9E-01	9.6E-01	9.9E-01	1.0E+00	1.0E+00
				5.50	7.2E-01	9.7E-01	1.0E+00	1.0E+00	1.0E+00
				5.75	7.5E-01	9.8E-01	1.0E+00	1.0E+00	1.0E+00
				6.00	7.7E-01	9.8E-01	1.0E+00	1.0E+00	1.0E+00
AIR HANDLING UNITS	6.90	0.27	0.45	0.50	3.6E-16	7.5E-12	1.2E-09	2.9E-08	2.8E-07
				1.00	7.5E-12	2.9E-08	1.6E-06	2.0E-05	1.2E-04
				1.25	1.3E-10	2.8E-07	1.2E-05	1.2E-04	5.7E-04
				1.50	1.2E-09	1.6E-06	5.2E-05	4.3E-04	1.8E-03
				1.75	6.7E-09	6.5E-06	1.7E-04	1.2E-03	4.5E-03
				2.00	2.9E-08	2.0E-05	4.3E-04	2.7E-03	9.1E-03
				2.25	9.3E-08	5.2E-05	9.4E-04	5.2E-03	1.6E-02
				2.50	2.8E-07	1.2E-04	1.8E-03	9.1E-03	2.7E-02
				2.75	7.2E-07	2.3E-04	3.2E-03	1.5E-02	4.0E-02
				3.00	1.6E-06	4.3E-04	5.2E-03	2.2E-02	5.6E-02
				3.25	3.4E-06	7.3E-04	8.0E-03	3.1E-02	7.3E-02
				3.50	6.5E-06	1.2E-03	1.2E-02	4.3E-02	9.8E-02
				3.75	1.2E-05	1.8E-03	1.6E-02	5.6E-02	1.2E-01
				4.00	2.0E-05	2.7E-03	2.2E-02	7.2E-02	1.5E-01
				4.25	3.3E-05	3.8E-03	2.9E-02	8.9E-02	1.8E-01
				4.50	5.2E-05	5.2E-03	3.7E-02	1.1E-01	2.1E-01
				4.75	7.9E-05	7.0E-03	4.6E-02	1.3E-01	2.4E-01
				5.00	1.2E-04	9.1E-03	5.6E-02	1.5E-01	2.7E-01
				5.25	1.7E-04	1.2E-02	6.8E-02	1.7E-01	3.0E-01
				5.50	2.3E-04	1.5E-02	8.0E-02	2.0E-01	3.3E-01
				5.75	3.2E-04	1.8E-02	9.3E-02	2.2E-01	3.6E-01
				6.00	4.3E-04	2.2E-02	1.1E-01	2.4E-01	3.9E-01

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COMPONENT	MEDIAN FRAGIL	LOG SD FRAGIL	LOG SD RESP	MULT FACTOR	1 SSE	SEISMIC FAILURE 2 SSE	PROBABILITY 3 SSE	4 SSE	5 SSE
GAS TURBINE GENERATOR	12.10	0.27	0.25	0.50	0.0E+00	3.6E-29	4.7E-24	9.7E-21	2.4E-18
				1.00	3.6E-29	9.7E-21	1.6E-16	7.8E-14	6.2E-12
				1.25	2.7E-26	2.4E-18	2.1E-14	6.2E-12	3.4E-10
				1.50	4.7E-24	1.6E-16	8.2E-13	1.7E-10	7.0E-09
				1.75	3.0E-22	4.8E-15	1.5E-11	2.3E-09	7.4E-08
				2.00	9.7E-21	7.8E-14	1.7E-10	1.9E-08	5.0E-07
				2.25	1.8E-19	8.2E-13	1.3E-09	1.1E-07	2.4E-06
				2.50	2.4E-18	6.2E-12	7.0E-09	5.0E-07	9.1E-06
				2.75	2.2E-17	3.6E-11	3.1E-08	1.8E-06	2.8E-05
				3.00	1.6E-16	1.7E-10	1.1E-07	5.5E-06	7.5E-05
				3.25	9.6E-16	6.7E-10	3.5E-07	1.5E-05	1.8E-04
				3.50	4.8E-15	2.3E-09	9.7E-07	3.5E-05	3.7E-04
				3.75	2.1E-14	7.0E-09	2.4E-06	7.5E-05	7.3E-04
				4.00	7.8E-14	1.9E-08	5.5E-06	1.5E-04	1.3E-03
				4.25	2.7E-13	4.8E-08	1.2E-05	2.8E-04	2.2E-03
				4.50	8.2E-13	1.1E-07	2.3E-05	4.9E-04	3.6E-03
				4.75	2.3E-12	2.4E-07	4.3E-05	8.2E-04	5.5E-03
				5.00	6.2E-12	5.0E-07	7.5E-05	1.3E-03	8.2E-03
				5.25	1.5E-11	9.7E-07	1.3E-04	2.0E-03	1.2E-02
				5.50	3.6E-11	1.8E-06	2.1E-04	3.0E-03	1.6E-02
				5.75	8.0E-11	3.2E-06	3.2E-04	4.3E-03	2.2E-02
				6.00	1.7E-10	5.5E-06	4.9E-04	6.0E-03	2.8E-02

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APPENDIX C.2

Review of Site-Specific Fragilities

Derived for 19 Recent PRAs



**STRUCTURAL
MECHANICS
ASSOCIATES**
A Calif. Corp.

NTS/SMA 12607.09

5160 Birch Street, Newport Beach, Calif. 92660 (714) 833-7552

January 21, 1985

Dr. M. P. Bohn
SANDIA NATIONAL LABORATORY
P.O. Box 5800
Albuquerque, New Mexico 87115

Dear Dr. Bohn:

We have completed a comprehensive review of the seismic fragility levels for mechanical and electrical equipment generally found in nuclear power plant facilities. The evaluation centered upon the 31 specific categories of equipment shown in Table 1. This letter report presents a description of the data base used to establish the median fragility levels and their variabilities, a discussion of the process used to obtain the final statistical parameters, and a discussion of the significant differences between the results of this study and fragility levels previously established during the SSMRP.

SMA has provided the fragility description for mechanical and electrical equipment in support of some 20 Probabilistic Risk Assessments (PRAs) of existing nuclear power plant facilities during the past four years. The calculated fragility levels for these PRAs have been based upon either plant specific stress levels obtained from code analysis reports, SQRT forms, and/or qualification test reports or upon a generic development using either knowledge of general equipment design practices or data from generic equipment dynamic tests. For the current study, all those components whose fragility was based upon well defined plant specific design stress information or dynamic qualification test data were included in the data base. All such components were placed in one of the defined categories and their floor level seismic acceleration capacity was computed based upon a recognition of the conservatism or unconservatism present in the seismic analysis of the component and the actual failure point of the component. The data base which can be obtained from plant specific seismic analyses is still somewhat limited such that the number of samples in each category ranges from 2 to 51. However, since the emphasis in the PRA evaluations involves a search for "weak links" or low capacity components within a safety related system, it is believed that the data base is at least slightly conservatively biased.

Table 1 presents the statistical parameters associated with the floor level seismic acceleration, capacities determined for the components in each category. The calculated median capacity values for each component

were used in the statistical evaluation. The results presented in Table 1 are defined as follows:

Sa_{Min} = Minimum median acceleration capacity value for all samples within the category

Sa_{Max} = Maximum median acceleration capacity value for all samples within the category

$\bar{Sa}$ = Mean value of acceleration capacity

σ = Standard deviation

δ = Coefficient of variation ($\delta = \sigma/\bar{Sa}$)

B_U = Uncertainty variability of the sample distribution

$$B_U = \sqrt{\ln(1 + \delta^2)}$$

B_R = Randomness variability of the sample distribution

$$B_R = \frac{\sum_{i=1}^N (B_{R_i})}{N}$$

$\bar{S}_a$ = Median value of acceleration capacity

$$\bar{S}_a = \bar{Sa} e^{-1/2 B_U^2}$$

HCLF = High Confidence - Low Frequency capacity value representing 95% confidence of less than 5% frequency of failure

$$HCLF = \bar{S}_a e^{-1.65 (B_R + B_U)}$$

The statistical median values and their variabilities were originally computed from the equations given above for all samples within each category. However, in many cases the uncertainty variability (B_U) was found to be inordinately large. By plotting the distribution of seismic acceleration capacity for each category, as shown in Figure 1 for vertical and horizontal pumps, it was found that high capacity outliers seriously distorted the distributions due to the limited number of samples in each category. Since it is of interest in this study to define the lower half of the fragility curve as accurately as possible (the upper half is not of concern), and

since a possibility exists that the data for these high capacity samples did not define the lowest capacity failure mode for the item, the extremely high capacity outliers were removed from the sample set and the statistical median values and their variabilities were recomputed. The values shown in Table 1 reflect results of the latter analysis. The high confidence-low frequency value calculated for each category was then compared with the experience data of similar components subjected to real earthquakes gathered by the Seismic Qualification Utility Group (SQUG). It was found in all cases that the calculated HCLF values exceeded earthquake accelerations at which no failures were found to occur. It was also judged that the resulting median capacity values ($\bar{Y}_a$) were realistic in magnitude and bore reasonable comparisons between the various categories. It should be noted that the uncertainty variability (β_u) given in Table 1 is based upon the variation of the median seismic capacity values of the samples in each category and is not related to the uncertainty in the calculation of the median values for each component. As a result, where the sample size is small (i.e., Safety Relief Valves or Compressors) or where the range from the minimum to maximum value is small (i.e., Motor Control Center -Trip Failure) the tabulated β_u values should probably be increased.

A comparison was made between the results of this study and the median seismic capacities recommended for the various categories from the SSMRP effort. In most cases there was reasonable agreement. However, there were a few notable exceptions which will be discussed. The median values and uncertainty variabilities from this study and the SSMRP recommended values are summarized in Table 2.

SSMRP recommended a seismic capacity for diesel generators of 0.65g. During the course of the PRA studies the diesel generators were reviewed in detail for most plants. The capacities were found to generally be well in excess of 1.0g. The results of this study indicate that a median capacity on the order of 1.95g is appropriate for diesel generators.

SSMRP recommended acceleration capacities of 1.84g and 2.01g for small tanks or accumulators and flat bottom storage tanks, respectively. In general small accumulators exhibit very high capacities in excess of 10g. This study indicated a capacity of 17.08g, which may be somewhat high. However, it should be noted that a very high capacity outlier was eliminated from the data set even to achieve the tabulated value. In contrast flat bottom storage tanks which are typically located outside have seldom been found to exhibit an acceleration capacity in excess of 1.0g. It is believed that the SSMRP value should be decreased significantly. The sample set from this study indicates that a median capacity of 0.45g is appropriate for flat bottom storage tanks.

SSMRP recommended acceleration capacities of 9.84g and 4.83g for small motor operated valves (≤ 4 inch) and large motor operated valves (>4 inch), respectively. It is believed that these should essentially be reversed. Numerous studies have shown that the large motor operators are more critical for small valves whose section modulus is small. This study also shows this trend and indicates that seismic capacities of 4.68g and 11.04g are appropriate for small and large motor operated valves, respectively.

Although the sample is small, the seismic capacity of inverters is indicated to be on the order of 2.01g in contrast to the 15.6g value recommended by SSMRP. It is believed that the SSMRP value was based upon a beefed up unit specifically designed for use in a high seismic plant.

Finally, more recent data indicates that the median seismic capacity of ceramic insulators should be increased to 0.30g over the 0.20g originally specified by the SSMRP.

Some care should be taken in the interpretation and use of the results of this study. It should be noted that median capacity values for a wide range of component designs are potentially included in this data set. As a result, the plant risk for a given nuclear facility may be governed by a component whose capacity is less than the median value for the appropriate category. Therefore, for initial screening studies employed to determine those components whose failure would be the most significant contributors to plant risk, it is recommended that a seismic capacity value somewhat less than the median value be specified. For example one might assume a median value equal to the minus one log standard deviation from the data base.

$$S_a = \bar{S}_a e^{-\beta_U}$$

However, if such a reduced value of seismic capacity is specified, the uncertainty variability (β_U) should also be reduced accordingly such that the HCLF of the data base would be unchanged. The reduced uncertainty variability would then be calculated as

$$\beta_U' = -\frac{1}{1.65} \left[\ln \left(\frac{\text{HCLF}}{S_a} \right) \right] - \beta_R$$

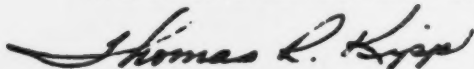
where S_a is the specified capacity value. Such an approach could identify those components which should be evaluated in more detail to assure acceptably low plant risk.

Dr. M. P. Gohn
SANDIA NATIONAL LABORATORY
January 21, 1985
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If you have any questions concerning the results of this study, please do not hesitate to give me a call at the number listed above. We thank you for the opportunity to be of service to you.

Sincerely,

NTS/STRUCTURAL MECHANICS ASSOCIATES



Thomas R. Kipp
Project Manager

cc: J. J. Johnson

VERTICAL PUMPS (15 Samples)



HORIZONTAL PUMPS (51 Samples)

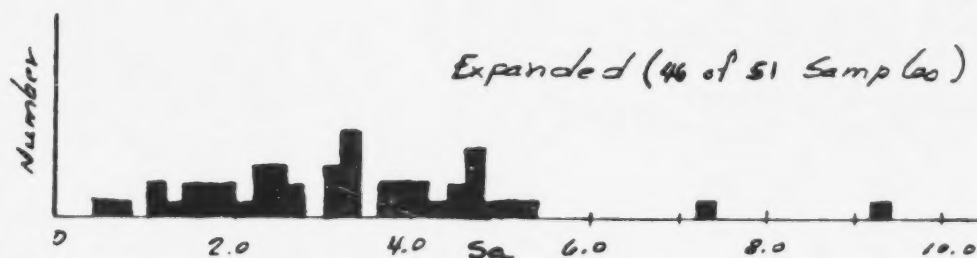
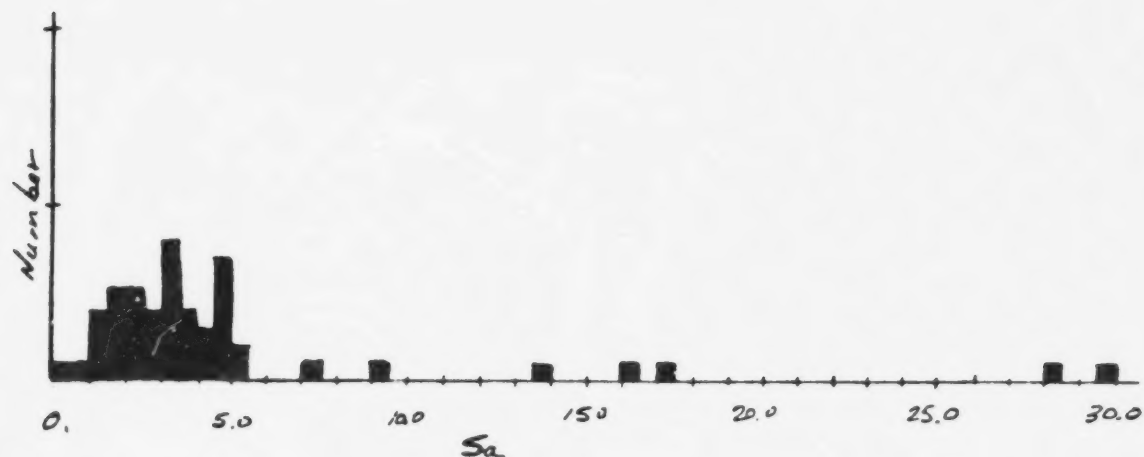


Figure 1 Seismic Capacity Distributions For Vertical and Horizontal Pumps

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TABLE 1 CAPACITIES AND VARIABILITIES FOR PLANT SPECIFIC EQUIPMENT CATEGORIES

CATEGORY	NO. OF SAMPLES	FREQUENCY RANGE	S _a MIN	S _a MAX	S _a	σ	δ	$\bar{S}_a$	B _U	B _R
Main Steam ISO Valve	5	5 - 20 Hz	3.02	10.80	7.15	3.00	0.42	6.59	0.40	0.24
Diesel Generator	10	11 - Rigid	0.75	3.74	2.13	0.95	0.45	1.95	0.43	0.16
Pumps (Long Column)	7	7 - Rigid	0.97	5.82	2.98	1.81	0.61	2.55	0.56	0.17
Pumps (Vertical)	14	7 - Rigid	0.98	5.67	3.29	1.56	0.48	2.97	0.45	0.21
Pumps (Horizontal)	44	7 - Rigid	0.49	5.37	3.03	1.29	0.43	2.79	0.41	0.15
Heat Exchangers (Vertical)	9	6 - Rigid	1.28	4.80	2.41	1.02	0.42	2.24	0.40	0.21
Heat Exchangers (Horizontal)	11	14 - Rigid	1.11	9.40	4.55	3.20	0.70	3.72	0.63	0.16
Tanks (Small Accumulators)	4	22 - Rigid	12.5	28.6	18.31	7.09	0.39	17.08	0.37	0.19
Tanks (Flat Bottom)	11	5 - 15	0.16	0.88	0.52	0.29	0.56	0.45	0.52	0.26
Tanks (Vertical)	13	3 - Rigid	1.44	7.50	3.66	1.90	0.54	3.21	0.51	0.15
Tanks (Horizontal)	14	3 - Rigid	1.22	6.56	3.35	1.60	0.46	3.02	0.45	0.19
Fan Coolers	8	4 - 22	1.86	16.80	0.30	5.56	0.67	6.90	0.61	0.25
Valves (Safety Relief)	4	5 - 20(Piping)	7.30	9.17	8.48	0.86	0.10	8.44	0.10	0.27
Valves (A.O & H.O $\leq 4"$)	9	5 - 20(Piping)	1.73	9.97	5.33	2.90	0.54	4.68	0.51	0.19
Valves (A.O & H.O $> 4"$)	47	5 - 20(Piping)	2.31	27.0	12.49	6.59	0.53	11.04	0.50	0.21
Compressors	2	10 - 20	2.36	3.09	2.73	0.52	0.19	2.68	0.19	0.12
Turbines (HPCI & RCIC)	6	15 - Rigid	2.92	6.09	5.21	1.22	0.23	5.10	0.23	0.19
Electrical Panels & Boards (Chatter)	41	2 - 20	1.60	10.23	4.71	2.50	0.53	4.19	0.50	0.19
Electrical Panels & Boards (Trip)	10	2 - 20	5.54	13.53	10.53	4.72	0.45	9.60	0.43	0.25
Switchgear (Chatter)	11	4 - 10	1.94	7.35	5.13	1.94	0.38	4.79	0.37	0.17
Switchgear (Trip)	6	4 - 10	4.58	9.98	6.73	1.98	0.29	6.43	0.29	0.21
Switchgear (Structural)	11	4 - 10	1.50	14.7	6.29	5.13	0.82	4.87	0.71	0.24
Motor Control Centers (Chatter)	11	4 - 18	1.79	8.53	4.83	2.34	0.49	4.35	0.46	0.16
Motor Control Centers (Trip)	5	4 - 10	6.85	8.00	7.29	0.45	0.06	7.27	0.06	0.21
Batteries and Racks	10	3 - Rigid	1.21	5.60	2.98	1.43	0.48	2.66	0.46	0.23
Battery Chargers	6	5 - 20	1.76	9.35	3.68	1.70	0.46	3.34	0.44	0.24
Transformers	6	5 - 20	5.49	14.25	9.32	3.34	0.36	8.77	0.35	0.26
Switches	4	13 - 20	1.73	6.46	3.43	2.26	0.65	2.92	0.59	0.25
Inverters	2	8 - 14	1.67	2.52	2.10	0.60	0.29	2.01	0.26	0.14
Instrumentation	5	Rigid	6.84	14.97	11.23	3.83	0.34	10.63	0.33	0.34
Ceramic Insulators	-	-	-	-	-	-	-	0.30	0.37	0.30

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TABLE 2 COMPARISON OF RESULTS WITH SSMRP RECOMMENDATIONS

CATEGORY	Current Study		SSMRP	
	γ_{Sa}	β_U	γ_{Sa}	β_U
Main Steam 150 Valve	6.59	0.40	-	-
Diesel Generator	1.95	0.43	0.65	0.40
Pumps (Long Column)	2.55	0.56	-	-
Pumps (Vertical)	2.97	0.45	2.21	0.39
Pumps (Horizontal)	2.79	0.41	3.19	0.34
Heat Exchangers (Vertical)	2.24	0.40	-	-
Heat Exchangers (Horizontal)	3.72	0.63	-	-
Tanks (Small Accumulators)	17.08	0.37	1.84	0.51
Tanks (Flat Bottom)	0.45	0.52	2.01	0.38
Tanks (Vertical)	3.21	0.51	1.46	0.56
Tanks (Horizontal)	3.02	0.45	3.91	0.61
Fan Coolers	6.90	0.61	2.24	0.41
Valves (Safety Relief)	8.44	0.10	8.90	0.40
Valves (A.O & M.O $\leq 4"$)	4.68	0.51	9.84	0.65
Valves (A.O & M.O $> 4"$)	11.04	0.50	4.83	0.65
Compressors	2.68	0.19	3.19	0.34
Turbines (HPCI & RCIC)	5.10	0.23	-	-
Electrical Panels & Boards (Chatter)	4.19	0.50	4.00	0.89
Electrical Panels & Boards (Trip)	9.60	0.43	7.63	0.88
Switchgear (Chatter)	4.79	0.37	2.33	0.81
Switchgear (Trip)	6.43	0.29	-	-
Switchgear (Structural)	4.87	0.71	11.50	0.88
Motor Control Centers (Chatter)	4.35	0.46	-	-
Motor Control Centers (Trip)	7.27	0.06	7.63	0.88
Batteries and Racks	2.68	0.46	2.29	0.50
Battery Chargers	3.34	0.44	-	-
Transformers	8.77	0.35	2.78	0.41
Switches	2.92	0.55	-	-
Inverters	2.01	0.28	15.60	0.44
Instrumentation	10.63	0.33	7.66	0.40
Ceramic Insulators	0.30	0.37	0.20	-

APPENDIX C.3

**Plant Specific Failure
Probabilities for Base Case**

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VARIABLE	VALUE
LPR1405A-VCC-LF	8.00E-03
LPI1401A-VCC-LF	8.00E-03
RBI2400B-VCC-LF	8.00E-03
SWS3821-VCC-LF	8.00E-03
SWS3810B-VCC-LF	8.00E-03
SWS3803-VCC-LF	8.00E-03
SWS001CX-COC-LF	3.60E-05
EFW0006X-CCC-LF	1.00E-04
EFW0013BX-CCC-LF	1.00E-04
EFW0002X-CCC-LF	1.00E-04
EFW0003X-CCC-LF	1.00E-04
EFW0010BX-CCC-LF	1.00E-04
EFW0099X-CCC-LF	1.00E-04
EFW0262X-CCC-LF	1.00E-04
HPIU348X-CCC-LF	1.00E-04
HPI1212X-CCC-LF	1.00E-04
HPI1215X-CCC-LF	1.00E-04
HPI1214X-CCC-LF	1.00E-04
HPIU19CX-CCC-LF	1.00E-04
HPIU34DX-CCC-LF	1.00E-04
LPIDH2AX-CCC-LF	1.00E-04
LPIDH14B-CCC-LF	1.00E-04
LPIDH17-CCC-LF	1.00E-04
LPIDH18-CCC-LF	1.00E-04
LPIDW48X-CCC-LF	1.00E-04
RBI9S4BX-CCC-LF	1.00E-04
RBIBW6AX-CCC-LF	1.00E-04
EFW00Y3X-NOC-LF	3.60E-05
LPI1428A-NOC-LF	3.60E-05
SWS3805-NCC-LF	8.00E-03
SWS3804-NCC-LF	8.00E-03
SWS3642B-V00-LF	3.00E-03
SWS3643A-V00-LF	3.00E-03
HPIU20BX-XOC-UTM	1.10E-04
HPIU20AX-XOC-UTM	1.10E-04
HPI1224X-XOC-UTM	1.10E-04
HPIU45DX-XOC-UTM	1.10E-04
HPIU18CX-XOC-UTM	1.10E-04
HPIU45AX-XOC-UTM	1.10E-04
HPIU45BX-XOC-UTM	1.10E-04
HPI1232X-XOC-UTM	1.10E-04
HPI1231X-XOC-UTM	1.10E-04
HPIMU25X-XOC-UTM	1.10E-04
HPIMU17X-XOC-UTM	1.10E-04
LPIDH3AX-XOC-UTM	1.10E-04
LPIDW8PX-XOC-UTM	1.10E-04
RBIBS1BX-XOC-UTM	1.10E-04

VARIABLE	VALUE
LPR1406B-VCC-LF	8.00E-03
LPI1400B-VCC-LF	8.00E-03
RBI2401A-VCC-LF	8.00E-03
SWS3802-VCC-LF	8.00E-03
SWS3808A-VCC-LF	8.00E-03
SWS3822-VCC-LF	8.00E-03
SWS001BX-COC-LF	3.60E-05
EFW013AX-CCC-LF	1.00E-04
EFW00Q1X-CCC-LF	1.00E-04
EFW00Q4X-CCC-LF	1.00E-04
EFW010AX-CCC-LF	1.00E-04
EFW0098X-CCC-LF	1.00E-04
EFW0261X-CCC-LF	1.00E-04
EFW00Q5X-CCC-LF	1.00E-04
HPIU34AX-CCC-LF	1.00E-04
HPIU34CX-CCC-LF	1.00E-04
HPI1213X-CCC-LF	1.00E-04
HPI1211X-CCC-LF	1.00E-04
HPIU19AX-CCC-LF	1.00E-04
HPIU19BX-CCC-LF	1.00E-04
LPIDH2BX-CCC-LF	1.00E-04
LPIDH14A-CCC-LF	1.00E-04
LPIDH13B-CCC-LF	1.00E-04
LPIDH13A-CCC-LF	1.00E-04
LPIBW4AX-CCC-LF	1.00E-04
RBIBS4AX-CCC-LF	1.00E-04
RBIBW6BX-CCC-LF	1.00E-04
EFW00Y4X-NOC-LF	3.60E-05
LPI1429B-NOC-LF	3.60E-05
SWS3841-NCC-LF	8.00E-03
SWS3840-NCC-LF	8.00E-03
SWS3640B-V00-LF	3.00E-03
SWS3820A-V00-LF	3.00E-03
HPIU18BX-XOC-UTM	1.10E-04
HPIU18AX-XOC-UTM	1.10E-04
HPI1223X-XOC-UTM	1.10E-04
HPIU20CX-XOC-UTM	1.10E-04
HPIMU26X-XOC-UTM	1.10E-04
HPI1234X-XOC-UTM	1.10E-04
HPI1233X-XOC-UTM	1.10E-04
HPIU45CX-XOC-UTM	1.10E-04
HPIOBW2X-XOC-UTM	1.10E-04
HPIMU16X-XOC-UTM	1.10E-04
HPIOBW3X-XOC-UTM	1.10E-04
LPIDH3BX-XOC-UTM	1.10E-04
RBIBW5BX-XOC-UTM	1.10E-04
RBIBW5AX-XOC-UTM	1.10E-04

• • • • VALUE BLOCK TABLE • • • •
(ANOSMVL - LEVEL 1)

VARIABLE	VALUE	VARIABLE	VALUE
RBIBS1AX-XOC-UTM	1.10E-04	SWS002CX-XOC-UTM	1.10E-04
SWS002BX-XOC-UTM	1.10E-04	SWS022BX-XOC-UTM	1.10E-04
SWS024CX-XOC-UTM	1.10E-04	SWS0218X-XOC-UTM	1.10E-04
SWS0061X-XOC-UTM	1.10E-04	SWS0064X-XOC-UTM	1.10E-04
SWS038AX-XOC-UTM	1.10E-04	SWS018CX-XOC-UTM	1.10E-04
SWS037CX-XOC-UTM	1.10E-04	SWS018AX-XOC-UTM	1.10E-04
SWS037AX-XOC-UTM	1.10E-04	SWS018BX-XOC-UTM	1.10E-04
SWS0015X-XOC-UTM	1.10E-04	SWS0014X-XOC-UTM	1.10E-04
SWS037BX-XOC-UTM	1.10E-04	SWS020AX-XOC-UTM	1.10E-04
SWS022AX-XOC-UTM	1.10E-04	SWS021AX-XOC-UTM	1.10E-04
SWS0063X-XOC-UTM	1.10E-04	SWS0086X-XOC-UTM	1.10E-04
SWS038BX-XOC-UTM	1.10E-04	SWS068X-XOC-UTM	1.10E-04
SWS067X-XOC-UTM	1.10E-04	SWS066X-XOC-UTM	1.10E-04
LPIDH8AX-XCO-UTM	1.10E-04	LPIDH8BX-XCO-UTM	1.10E-04
RBIDH10X-XCO-UTM	1.10E-04	HPRMU24X-XCC-UTM	3.00E-05
HPRMU23X-XCC-UTM	3.00E-05	HPRMU14X-XCC-UTM	3.00E-05
HPRMU15X-XCC-UTM	3.00E-05	HPRDH7BX-XCC-UTM	3.00E-05
HPRDH7AX-XCC-UTM	3.00E-05	EFW00X2B-VOC-UTM	1.10E-04
EFW2670A-VOC-UTM	1.10E-04	EFW2627A-VOC-UTM	1.10E-04
EFW2620P-VOC-UTM	1.10E-04	EFW2626B-VOC-UTM	1.10E-04
EFW2802B-VOC-UTM	1.10E-04	EFW2667A-VOC-UTM	1.10E-04
EFW2666B-VOC-UTM	1.10E-04	EFW2617B-VOC-UTM	1.10E-04
EFW00X3A-VOC-UTM	1.10E-04	EFW00X4B-VOC-UTM	1.10E-04
EFW00X1A-VOC-UTM	1.10E-04	EFW2800A-VOC-UTM	1.10E-04
LPR1414A-VOC-UTM	1.10E-04	LPR1415B-VOC-UTM	1.10E-04
SWS3641B-VOC-UTM	1.10E-04	SWS3644A-VOC-UTM	1.10E-04
SWS3646A-VOC-UTM	1.10E-04	SWS3645A-VOC-UTM	1.10E-04
SWS3809A-VOC-UTM	1.10E-04	EFW00Y1A-VCC-UTM	3.00E-05
EFW00Y2B-VCC-UTM	3.00E-05	SWS3806A-VCC-UTM	3.00E-05
SWS3807B-VCC-UTM	3.00E-05	LPI1407A-VCC-UTM	3.00E-05
LPI1408B-VCC-UTM	3.00E-05	HPI1220A-VCC-UTM	3.00E-05
HPI1219A-VCC-UTM	3.00E-05	HPI1228B-VCC-UTM	3.00E-05
HPI1227B-VCC-UTM	3.00E-05	LPR1405A-VCC-UTM	3.00E-05
LPR1406B-VCC-UTM	3.00E-05	RBI2400B-VCC-UTM	3.00E-05
RBI2401A-VCC-UTM	3.00E-05	SWS3821-VCC-UTM	3.00E-05
SWS3802-VCC-UTM	3.00E-05	SWS3810B-VCC-UTM	3.00E-05
SWS3808A-VCC-UTM	3.00E-05	SWS3803-VCC-UTM	3.00E-05
SWS3822-VCC-UTM	3.00E-05	LPI1400B-VCC-UTM	3.00E-05
LPI1401A-VCC-UTM	3.00E-05	EFW00Y3X-NOC-UTM	1.10E-04
EFW00Y4X-NOC-UTM	1.10E-04	LPI1428A-NOC-UTM	1.10E-04
LPI1429B-NOC-UTM	1.10E-04	SWS3805-NCC-UTM	1.10E-04
SWS3841-NCC-UTM	1.10E-04	SWS3804-NCC-UTM	1.10E-04
SWS3840-NCC-UTM	1.10E-04	SWS3642B-VOC-UTM	3.00E-05
SWS3640B-VOC-UTM	3.00E-05	SWS3643A-VOC-UTM	3.00E-05
SWS3820A-VOC-UTM	3.00E-05	SRV-1-LF	1.00E-05
SRV-2-LF	1.00E-05	ERV-LF	3.00E-04
BLKVLV-LF	8.00E-03	BLKVLV-GATE	1.00E+00

* * * * VALUE BLOCK TABLE * * * *
(ANOSMVL - LEVEL 1)

VARIABLE	VALUE
EFWOP7BA-PMD-LF	8.70E-03
HPIP36CB-PMD-LF	8.70E-03
HPIP36AA-PMD-LF	8.70E-03
LPIOP34B-PMD-LF	8.70E-03
RBIP35AA-PMD-LF	8.70E-03
SWSOP4BA-PMD-LF	8.70E-03
SWSOP4BA-PAC-LF	8.70E-03
HPIP36BA-PMD-UTM	2.80E-04
HPIP36CB-PMD-UTM	2.80E-04
LPIOP34B-PMD-UTM	2.80E-04
RBIP35AA-PMD-UTM	2.80E-04
SWSOP4BA-PMD-UTM	0.
SWSOP4BA-PAC-UTM	0.
DG-CM	8.90E-03
EFWSP-CM	9.10E-04
RBSIP-CM	8.80E-05
SWSP-CM	6.50E-03
HPSV-CM	8.20E-04
RBC-SWL-CM	1.50E-03
LPSV-CM	1.10E-03
RBSIV-CM	1.10E-03
SWSV-40-41-CM	1.10E-03
SWSV-21-22-CM	1.10E-03
SWSV-06-07-CM	1.10E-03
CSA-CM	4.70E-04
A1-4160VAC-LF	5.04E-01
B5-480VAC-LF	5.04E-01
A3-4160VAC-LF	5.04E-01
B55-480VAC-LF	5.04E-01
B51-480VAC-LF	5.04E-01
B61-480VAC-LF	5.04E-01
RA1-125VDC-LF	3.00E-05
D11-125VDC-LF	3.00E-05
D02-125VDC-LF	3.00E-05
LF-DC-RS1	3.00E-05
B55-480VAC-UTM	0.
B5-480VAC-UTM	0.
B51-480VAC-UTM	0.
A2-4160VAC-UTM	0.
A4-4160VAC-UTM	0.
B62-480VAC-UTM	0.
RA1-125VDC-UTM	0.
D11-125VDC-UTM	0.
D21-125VDC-UTM	0.
TM-DC-RS1	0.
DG1-GEN-LF	3.80E-02
DG1-GEN-UTM	6.10E-03

VARIABLE	VALUE
EFWOP7AX-PTD-LF	3.50E-02
HPIP36BA-PMD-LF	8.70E-03
LPIOP34A-PMD-LF	8.70E-03
RBIP35BB-PMD-LF	8.70E-03
SWSOP4CB-PMD-LF	8.70E-03
SWSOP4CB-PAC-LF	8.70E-03
EFWOP7BA-PMD-UTM	2.80E-04
HPIP36AA-PMD-UTM	2.80E-04
LPIOP34A-PMD-UTM	2.80E-04
RBIP35BB-PMD-UTM	2.80E-04
SWSOP4CB-PMD-UTM	0.
SWSOP4CB-PAC-UTM	0.
EFWOP7AX-PTD-UTM	6.80E-04
BATT-CM	5.20E-04
LPSP-CM	8.80E-05
HPSP-CM	8.40E-05
SRV-CM	1.30E-06
EFWSV-CM	8.20E-04
HPISV-CM	1.10E-03
LPRSV-CM	1.10E-03
SWSV-08-10-CM	1.10E-03
SWSV-02-03-CM	1.10E-03
SWSV-04-05-CM	1.10E-03
SWSV-40-42-CM	1.10E-03
CSB-CM	4.70E-04
A2-4160VAC-LF	5.04E-01
B6-480VAC-LF	5.04E-01
A4-4160VAC-LF	5.04E-01
B56-480VAC-LF	5.04E-01
B52-480VAC-LF	5.04E-01
B62-480VAC-LF	5.04E-01
RA2-125VDC-LF	3.00E-05
D21-125VDC-LF	3.00E-05
D01-125VDC-LF	1.00E+00
LF-DC-RS2	3.00E-05
B56-480VAC-UTM	0.
A3-4160VAC-UTM	0.
B52-480VAC-UTM	0.
B6-480VAC-UTM	0.
B61-480VAC-UTM	0.
A1-4160VAC-UTM	0.
RA2-125VDC-UTM	0.
D01-125VDC-UTM	0.
D02-125VDC-UTM	0.
TM-DC-RS2	0.
DG2-GEN-LF	3.80E-02
DG2-GEN-UTM	6.10E-03

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• • • • VALUE BLOCK TABLE • • • •
(ANOSMVL - LEVEL 1)

VARIABLE	VALUE	VARIABLE	VALUE
LF-DC-Y11AC	3.70E-04	LF-DC-Y22AC	3.70E-04
LF-DC-Y11	2.60E-03	LF-DC-Y22	2.60E-03
LF-DC-Y11DC	3.40E-04	LF-DC-Y22DC	3.40E-04
LF-DC-RS1ALTAC	2.70E-04	LF-DC-RS2ALTAC	2.70E-04
BATTD07-ZBT-LF	1.80E-03	BATTD06-ZBT-LF	1.80E-03
BATTD07-ZBT-UTM	1.10E-03	BATTD06-ZBT-UTM	1.10E-03
D03-CHARGER-LF	5.40E-05	D04-CHARGER-LF	5.40E-05
LOSS-OSP	1.00E+00	TM-LOGIC-CM	2.00E-04
ACT-SG-SENS-CM	3.00E-05	ACT-EFAA-LF	5.00E-03
ACT-EFBB-LF	5.00E-03	ACT-EFXA-LF	5.00E-03
ACT-EFXB-LF	5.00E-03	LF-REF-RM95	5.50E-03
LF-REF-RM109	5.50E-03	LF-ECS-A1	1.00E-04
LF-ECS-B1	1.00E-04	LF-ECS-A2	1.00E-04
LF-ECS-B2	1.00E-04	LF-ECS-A4	1.00E-04
LF-ECS-B4	1.00E-04	LF-ECS-A5	3.80E-03
LF-ECS-B5	4.80E-03	LF-ECS-A6	4.80E-03
LF-ECS-B6	6.40E-03	DIG-LOG-CM-TM	1.70E-06
LF-ESF-T1	1.90E-05	LF-ESF-T2	1.90E-05
LF-ESF-T3	1.90E-05	LF-ESF-T4	1.90E-05
LF-ESF-T5	1.90E-05	LF-ESF-T6	1.90E-05
LF-ESF-T7	1.90E-05	LF-ESF-T8	1.90E-05
LF-ESF-TRS1	4.00E-05	LF-ESF-TRS2	4.00E-05
ESFA110X-LOG-LF	1.00E-04	ESFA210X-LOG-LF	1.00E-04
ESFA310X-LOG-LF	1.00E-04	ESFA116X-LOG-LF	1.00E-04
ESFA216X-LOG-LF	1.00E-04	ESFA316X-LOG-LF	1.00E-04
ESFA122-LOG-LF	1.00E-04	ESFA222-LOG-LF	1.00E-04
ESFA322-LOG-LF	1.00E-04	ESFA123-LOG-LF	1.00E-04
ESFA223-LOG-LF	1.00E-04	ESFA323-LOG-LF	1.00E-04
LF-ESF-TA01	1.20E-02	LF-ESF-TB01	1.20E-02
LF-ESF-TC01	1.20E-02	LF-ESF-TA120	1.20E-02
LF-ESF-TA220	1.20E-02	LF-ESF-TA320	1.20E-02
LF-ESF-TA119	1.20E-02	LF-ESF-TA219	1.20E-02
LF-ESF-TA319	1.20E-02	LF-ESF-TA04	4.50E-04
LF-ESF-TB04	4.50E-04	LF-ESF-TC04	4.50E-04
BISTABLEX08-CM	1.30E-03	BISTABLEX19-CM	1.30E-03
BISTABLEX20-CM	1.30E-03	ESF-TSL	0.
LF-SWS-VCH4A	1.90E-02	LF-SWS-VCH4B	1.90E-02
CST-TNK-LF	0.	BWST-TNK-LF	1.40E-03
HPRE35AX-HTX-LFB	3.00E-04	HPRE35BX-HTX-LFB	3.00E-04
HPRE35AX-HTX-UTM	6.00E-04	HPRE35BX-HTX-UTM	6.00E-04
LF-RBC-SEGC8	5.90E-02	LF-RBC-SEGAA	5.90E-02
LF-RBC-SEG8A	5.90E-02	LF-RBC-SEGDB	5.90E-02
LF-RBC-SWL1	2.30E-02	LF-RBC-SWL2	2.30E-02
FNB-MANACT	3.00E-03	OPF-OPEN-FNB	3.00E-03
HPRS-MANACT	3.00E-03	RBSR-MANACT	3.00E-03
LPRS-MANACT	3.00E-03	DG-MANACT	5.00E-02

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• • • • VALUE BLOCK TABLE • • • •
(ANOSMVL - LEVEL 2)

VARIABLE	VALUE	VARIABLE	VALUE
LPR1405A-VCC-LF	8.00E-03	LPR1406B-VCC-LF	8.00E-03
LPI1401A-VCC-LF	8.00E-03	LPI1400B-VCC-LF	8.00E-03
RBI2400B-VCC-LF	8.00E-03	RBI2401A-VCC-LF	8.00E-03
SWS3821-VCC-LF	8.00E-03	SWS3802-VCC-LF	8.00E-03
SWS3810B-VCC-LF	8.00E-03	SWS3808A-VCC-LF	8.00E-03
SWS3803-VCC-LF	8.00E-03	SWS3822-VCC-LF	8.00E-03
SWS001CX-COC-LF	3.60E-05	SWS001BX-COC-LF	3.60E-05
EFW0006X-CCC-LF	1.00E-04	EFW013AX-CCC-LF	1.00E-04
EFW013BX-CCC-LF	1.00E-04	EFW00Q1X-CCC-LF	1.00E-04
EFW00Q2X-CCC-LF	1.00E-04	EFW00Q4X-CCC-LF	1.00E-04
EFW00Q3X-CCC-LF	1.00E-04	EFW010AX-CCC-LF	1.00E-04
EFW010BX-CCC-LF	1.00E-04	EFW0098X-CCC-LF	1.00E-04
EFW0099X-CCC-LF	1.00E-04	EFW0261X-CCC-LF	1.00E-04
EFW0262X-CCC-LF	1.00E-04	EFW00Q5X-CCC-LF	1.00E-04
HPIU34BX-CCC-LF	1.00E-04	HPIU34AX-CCC-LF	1.00E-04
HPI1212X-CCC-LF	1.00E-04	HPIU34CX-CCC-LF	1.00E-04
HPI1215X-CCC-LF	1.00E-04	HPI1213X-CCC-LF	1.00E-04
HPI1214X-CCC-LF	1.00E-04	HPI1211X-CCC-LF	1.00E-04
HPIU19CX-CCC-LF	1.00E-04	HPIU19AX-CCC-LF	1.00E-04
HPIU34DX-CCC-LF	1.00E-04	HPIU19BX-CCC-LF	1.00E-04
LPIDH2AX-CCC-LF	1.00E-04	LPIDH2BX-CCC-LF	1.00E-04
LPIDH14B-CCC-LF	1.00E-04	LPIDH14A-CCC-LF	1.00E-04
LPIDH17-CCC-LF	1.00E-04	LPIDH13B-CCC-LF	1.00E-04
LPIDH18-CCC-LF	1.00E-04	LPIDH13A-CCC-LF	1.00E-04
LPBW48X-CCC-LF	1.00E-04	LPBW4AX-CCC-LF	1.00E-04
RBIBS4BX-CCC-LF	1.00E-04	RBIBS4AX-CCC-LF	1.00E-04
RBIBW6AX-CCC-LF	1.00E-04	RBIBW6BX-CCC-LF	1.00E-04
EFW00Y3X-NOC-LF	8.90E-05	EFW00Y4X-NOC-LF	8.90E-05
LPI1428A-NOC-LF	8.90E-05	LPI1429B-NOC-LF	8.90E-05
SWS3805-NCC-LF	8.00E-03	SWS3841-NCC-LF	8.00E-03
SWS3804-NCC-LF	8.00E-03	SWS3840-NCC-LF	8.00E-03
SWS3642B-V00-LF	3.00E-03	SWS3640B-V00-LF	3.00E-03
SWS3643A-V00-LF	3.00E-03	SWS3820A-V00-LF	3.00E-03
HPIU20BX-XOC-UTM	1.10E-04	HPIU18BX-XOC-UTM	1.10E-04
HPIU20AX-XOC-UTM	1.10E-04	HPIU18AX-XOC-UTM	1.10E-04
HPI1224X-XOC-UTM	1.10E-04	HPI1223X-XOC-UTM	1.10E-04
HPIU45DX-XOC-UTM	1.10E-04	HPIU20CX-XOC-UTM	1.10E-04
HPIU18CX-XOC-UTM	1.10E-04	HPIU26X-XOC-UTM	1.10E-04
HPIU45AX-XOC-UTM	1.10E-04	HPI1234X-XOC-UTM	1.10E-04
HPIU45BX-XOC-UTM	1.10E-04	HPI1233X-XOC-UTM	1.10E-04
HPI1232X-XOC-UTM	1.10E-04	HPIU45CX-XOC-UTM	1.10E-04
HPI1231X-XOC-UTM	1.10E-04	HPIOBW2X-XOC-UTM	1.10E-04
HPIU25X-XOC-UTM	1.10E-04	HPIU16X-XOC-UTM	1.10E-04
HPIU17X-XOC-UTM	1.10E-04	HPIOBW3X-XOC-UTM	1.10E-04
LPIDH3AX-XOC-UTM	1.10E-04	LPIDH3BX-XOC-UTM	1.10E-04
LPBW8BX-XOC-UTM	1.10E-04	RBIBW5BX-XOC-UTM	1.10E-04
RBIBS1BX-XOC-UTM	1.10E-04	RBIBW5AX-XOC-UTM	1.10E-04

• • • • VALUE BLOCK TABLE • • • •
(ANOSMVL - LEVEL 2)

VARIABLE	VALUE	VARIABLE	VALUE
RBIBS1AX-XOC-UTM	1.10E-04	SWS002CX-XOC-UTM	1.10E-04
SWS002BX-XOC-UTM	1.10E-04	SWS022BX-XOC-UTM	1.10E-04
SWS024CX-XOC-UTM	1.10E-04	SWS0218X-XOC-UTM	1.10E-04
SWS0061X-XOC-UTM	1.10E-04	SWS0064X-XOC-UTM	1.10E-04
SWS038AX-XOC-UTM	1.10E-04	SWS018CX-XOC-UTM	1.10E-04
SWS037CX-XOC-UTM	1.10E-04	SWS018AX-XOC-UTM	1.10E-04
SWS037AX-XOC-UTM	1.10E-04	SWS018BX-XOC-UTM	1.10E-04
SWS0015X-XOC-UTM	1.10E-04	SWS0014X-XOC-UTM	1.10E-04
SWS037BX-XOC-UTM	1.10E-04	SWS020AX-XOC-UTM	1.10E-04
SWS022AX-XOC-UTM	1.10E-04	SWS021AX-XOC-UTM	1.10E-04
SWS0063X-XOC-UTM	1.10E-04	SWS0086X-XOC-UTM	1.10E-04
SWS038BX-XOC-UTM	1.10E-04	SWS068X-XOC-UTM	1.10E-04
SWS067X-XOC-UTM	1.10E-04	SWS066X-XOC-UTM	1.10E-04
LPIDH8AX-XCO-UTM	1.10E-04	LPIDH8BX-XCO-UTM	1.10E-04
RBIDH10X-XCO-UTM	1.10E-04	HPRMU24X-XCC-UTM	3.00E-05
HPRMU23X-XCC-UTM	3.00E-05	HPRMU14X-XCC-UTM	3.00E-05
HPRMU15X-XCC-UTM	3.00E-05	HPRDH7BX-XCC-UTM	3.00E-05
HPRDH7AX-XCC-UTM	3.00E-05	EFW00X2B-VOC-UTM	1.10E-04
EFW2670A-VOC-UTM	1.10E-04	EFW2627A-VOC-UTM	1.10E-04
EFW2620B-VOC-UTM	1.10E-04	EFW2626B-VOC-UTM	1.10E-04
EFW2802B-VOC-UTM	1.10E-04	EFW2667A-VOC-UTM	1.10E-04
EFW2665B-VOC-UTM	1.10E-04	EFW2617B-VOC-UTM	1.10E-04
EFW00X3A-VOC-UTM	1.10E-04	EFW00X4B-VOC-UTM	1.10E-04
EFW00X1A-VOC-UTM	1.10E-04	EFW2800A-VOC-UTM	1.10E-04
LPR1414A-VOC-UTM	1.10E-04	LPR1415B-VOC-UTM	1.10E-04
SWS3641B-VOC-UTM	1.10E-04	SWS3644A-VOC-UTM	1.10E-04
SWS3646A-VOC-UTM	1.10E-04	SWS3645A-VOC-UTM	1.10E-04
SWS3809A-VOC-UTM	1.10E-04	EFW00Y1A-VCC-UTM	3.00E-05
EFW00Y2B-VCC-UTM	3.00E-05	SWS3806A-VCC-UTM	3.00E-05
SWS3807B-VCC-UTM	3.00E-05	LPI1407A-VCC-UTM	3.00E-05
LPI1408B-VCC-UTM	3.00E-05	HPI1220A-VCC-UTM	3.00E-05
HPI1219A-VCC-UTM	3.00E-05	HPI1228B-VCC-UTM	3.00E-05
HPI1227B-VCC-UTM	3.00E-05	LPR1405A-VCC-UTM	3.00E-05
LPR1406B-VCC-UTM	3.00E-05	RBI2400B-VCC-UTM	3.00E-05
RBI2401A-VCC-UTM	3.00E-05	SWS3821-VCC-UTM	3.00E-05
SWS3802-VCC-UTM	3.00E-05	SWS3810B-VCC-UTM	3.00E-05
SWS3808A-VCC-UTM	3.00E-05	SWS3803-VCC-UTM	3.00E-05
SWS3822-VCC-UTM	3.00E-05	LPI1400B-VCC-UTM	3.00E-05
LPI1401A-VCC-UTM	3.00E-05	EFW00Y3X-NOC-UTM	1.10E-04
EFW00Y4X-NOC-UTM	1.10E-04	LPI1428A-NOC-UTM	1.10E-04
LPI1429B-NOC-UTM	1.10E-04	SWS3805-NCC-UTM	1.10E-04
SWS3841-NCC-UTM	1.10E-04	SWS3804-NCC-UTM	1.10E-04
SWS3840-NCC-UTM	1.10E-04	SWS3642B-VOC-UTM	3.00E-05
SWS3640B-VOC-UTM	3.00E-05	SWS3643A-VOC-UTM	3.00E-05
SWS3820A-VOC-UTM	3.00E-05	SRV-1-LF	1.00E-05
SRV-2-LF	1.00E-05	ERV-LF	3.00E-04
BLKVLV-LF	8.00E-03	BLKVLV-GATE	1.00E+00

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• • • • VALUE BLOCK TABLE • • • •
(ANOSMVL - LEVEL 2)

VARIABLE	VALUE	VARIABLE	VALUE
EFWOP7BA-PMD-LF	8.70E-03	EFWOP7AX-PTD-LF	3.50E-02
HPIP36CB-PMD-LF	8.70E-03	HPIP36BA-PMD-LF	8.70E-03
HPIP36AA-PMD-LF	8.70E-03	LPIOP34A-PMD-LF	8.70E-03
LPIOP34B-PMD-LF	8.70E-03	RBIP35BB-PMD-LF	8.70E-03
RBIP35AA-PMD-LF	8.70E-03	SWSOP4CB-PMD-LF	8.70E-03
SWSOP4BA-PMD-LF	8.70E-03	SWSOP4CB-PAC-LF	8.70E-03
SWSOP4BA-PAC-LF	8.70E-03	EFWOP7BA-PMD-UTM	2.80E-04
HPIP36BA-PMD-UTM	2.80E-04	HPIP36AA-PMD-UTM	2.80E-04
HPIP36CB-PMD-UTM	2.80E-04	LPIOP34A-PMD-UTM	2.80E-04
LPIOP34B-PMD-UTM	2.80E-04	RBIP35BB-PMD-UTM	2.80E-04
RBIP35AA-PMD-UTM	2.80E-04	SWSOP4CB-PMD-UTM	0.
SWSOP4BA-PMD-UTM	0.	SWSOP4CB-PAC-UTM	0.
SWSOP4BA-PAC-UTM	0.	EFWOP7AX-PTD-UTM	6.80E-04
DG-CM	8.90E-03	BATT-CM	9.50E-04
EFWSP-CM	9.10E-04	LPSP-CM	8.80E-05
RBSIP-CM	8.80E-05	HPSP-CM	8.40E-05
SWSP-CM	6.50E-03	SRV-CM	1.30E-06
HPSV-CM	8.20E-04	EFWSV-CM	8.20E-04
RBC-SWL-CM	1.50E-03	HPISV-CM	1.10E-03
LPSV-CM	1.10E-03	LPRSV-CM	1.10E-03
RBSIV-CM	1.10E-03	SWSV-08-10-CM	1.10E-03
SWSV-40-41-CM	1.10E-03	SWSV-02-03-CM	1.10E-03
SWSV-21-22-CM	1.10E-03	SWSV-04-05-CM	1.10E-03
SWSV-06-07-CM	1.10E-03	SWSV-49-42-CM	1.10E-03
CSA-CM	4.70E-04	CSB-CM	4.70E-04
A1-4160VAC-LF	9.05E-01	A2-4160VAC-LF	9.05E-01
B5-480VAC-LF	9.05E-01	B6-480VAC-LF	9.05E-01
A3-4160VAC-LF	9.05E-01	A4-4160VAC-LF	9.05E-01
B55-480VAC-LF	9.05E-01	B56-480VAC-LF	9.05E-01
B51-480VAC-LF	9.05E-01	B52-480VAC-LF	9.05E-01
B61-480VAC-LF	9.05E-01	B62-480VAC-LF	9.05E-01
RA1-125VDC-LF	3.90E-05	RA2-125VDC-LF	3.90E-05
D11-125VDC-LF	3.90E-05	D21-125VDC-LF	3.90E-05
D02-125VDC-LF	3.90E-05	D01-125VDC-LF	3.90E-05
LF-DC-RS1	3.90E-05	LF-DC-RS2	3.90E-05
B55-480VAC-UTM	0.	B56-480VAC-UTM	0.
P5-480VAC-UTM	0.	A3-4160VAC-UTM	0.
B51-480VAC-UTM	0.	B52-480VAC-UTM	0.
A2-4160VAC-UTM	0.	B6-480VAC-UTM	0.
A4-4160VAC-UTM	0.	B61-480VAC-UTM	0.
B62-480VAC-UTM	0.	A1-4160VAC-UTM	0.
RA1-125VDC-UTM	0.	RA2-125VDC-UTM	0.
D11-125VDC-UTM	0.	D01-125VDC-UTM	0.
D21-125VDC-UTM	0.	D02-125VDC-UTM	0.
TM-DC-RS1	0.	TM-DC-RS2	0.
DG1-GEN-LF	3.80E-02	DG2-GEN-LF	3.80E-02
DG1-GEN-UTM	6.10E-03	DG2-GEN-UTM	6.10E-03

* * * * VALUE BLOCK TABLE * * * *
(ANOSMVL - LEVEL 2)

VARIABLE	VALUE
LF-DC-Y11AC	1.50E-02
LF-DC-Y11	1.50E-02
LF-DC-Y11DC	1.50E-02
LF-DC-RS1ALTAC	1.50E-02
BATTD07-ZBT-LF	6.40E-03
BATTD07-ZBT-UTM	1.10E-03
D03-CHARGER-LF	1.60E-04
LOSS-OSP	1.00E+00
ACT-SG-SENS-CM	3.00E-05
ACT-EFBB-LF	5.20E-03
ACT-EFXB-LF	5.20E-03
LF-REF-RM109	5.50E-03
LF-ECS-B1	1.00E-04
LF-ECS-B2	1.00E-04
LF-ECS-B4	1.00E-04
LF-ECS-B5	4.80E-03
LF-ECS-B6	6.40E-03
LF-ESF-T1	2.30E-04
LF-ESF-T3	2.30E-04
LF-ESF-T5	2.30E-04
LF-ESF-T7	2.30E-04
LF-ESF-TRS1	2.50E-04
ESFA110X-LOG-LF	3.10E-04
ESFA310X-LOG-LF	3.10E-04
ESFA216X-LOG-LF	3.10E-04
ESFA122-LOG-LF	3.10E-04
ESFA322-LOG-LF	3.10E-04
ESFA223-LOG-LF	3.10E-04
LF-ESF-TA01	1.20E-02
LF-ESF-TC01	1.20E-02
LF-ESF-TA220	1.20E-02
LF-ESF-TA119	1.20E-02
LF-ESF-TA319	1.20E-02
LF-ESF-TB04	6.60E-04
BISTABLEX20-CM	1.30E-03
LF-SWS-VCH4A	1.90E-02
CST-TNK-LF	0.
HPRE35AX-HTX-LFB	3.00E-04
HPRE35AX-HTX-UTM	6.00E-04
LF-RBC-SEGB	5.00E-01
LF-RBC-SEGBA	5.00E-01
LF-RBC-SWL1	2.30E-02
FNB-MANACT	3.00E-03
HPRS-MANACT	3.00E-03
LPRS-MANACT	3.00E-03

VARIABLE	VALUE
LF-DC-Y22AC	1.50E-02
LF-DC-Y22	1.50E-02
LF-DC-Y22DC	1.50E-02
LF-DC-RS2ALTAC	1.50E-02
BATTD06-ZBT-LF	6.40E-03
BATTD06-ZBT-UTM	1.10E-03
D04-CHARGER-LF	1.60E-04
TM-LOGIC-CM	2.00E-04
ACT-EFAA-LF	5.20E-03
ACT-EFXA-LF	5.20E-03
LF-REF-RM95	5.50E-03
LF-ECS-A1	1.00E-04
LF-ECS-A2	1.00E-04
LF-ECS-A4	1.00E-04
LF-ECS-A5	3.80E-03
LF-ECS-A6	4.80E-03
DIG-LOG-CM-TM	1.70E-06
LF-ESF-T2	2.30E-04
LF-ESF-T4	2.30E-04
LF-ESF-T6	2.30E-04
LF-ESF-T8	2.30E-04
LF-ESF-TRS2	2.50E-04
ESFA210X-LOG-LF	3.10E-04
ESFA116X-LOG-LF	3.10E-04
ESFA316X-LOG-LF	3.10E-04
ESFA222-LOG-LF	3.10E-04
ESFA123-LOG-LF	3.10E-04
ESFA323-LOG-LF	3.10E-04
LF-ESF-TB01	1.20E-02
LF-ESF-TA120	1.20E-02
LF-ESF-TA320	1.20E-02
LF-ESF-TA219	1.20E-02
LF-ESF-TA04	6.60E-04
LF-ESF-TC04	6.60E-04
ESF-TSL	0.
LF-SWS-VCH4B	1.90E-02
BWST-TNK-LF	1.15E-01
HPRE35BX-HTX-LFB	3.00E-04
HPRE35BX-HTX-UTM	6.00E-04
LF-RBC-SEGAA	5.00E-01
LF-RBC-SEGDB	5.00E-01
LF-RBC-SWL2	2.30E-02
OPF-OPEN-FNB	3.00E-03
RBSR-MANACT	3.00E-03
DG-MANACT	5.00E-02

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• • • • VALUE BLOCK TABLE • • • •
(ANOSMVL - LEVEL 3)

VARIABLE	VALUE	VARIABLE	VALUE
EFW0019X-XOC-LF	6.70E-04	HPIU45AX-XOC-LF	6.70E-04
HPI1234X-XOC-LF	6.70E-04	HPIU45BX-XOC-LF	6.70E-04
HPI1233X-XOC-LF	6.70E-04	HPI1224X-XOC-LF	6.70E-04
HPI1223X-XOC-LF	6.70E-04	HPI1232X-XOC-LF	6.70E-04
HPIU45CX-XOC-LF	6.70E-04	HPI1231X-XOC-LF	6.70E-04
HPIU20CX-XOC-LF	6.70E-04	HPIU18CX-XOC-LF	6.70E-04
HPIOBW2X-XOC-LF	6.70E-04	HPIMU25X-XOC-LF	6.70E-04
HPIMU26X-XOC-LF	6.70E-04	HPIMU16X-XOC-LF	6.70E-04
HPIMU17X-XOC-LF	6.70E-04	HPIOBW3X-XOC-LF	6.70E-04
HPIU20AX-XOC-LF	6.70E-04	HPIU18AX-XOC-LF	6.70E-04
HPIU45DX-XOC-LF	6.70E-04	HPIU20BX-XOC-LF	6.70E-04
HPIU18BX-XOC-LF	6.70E-04	LPI DH3AX-XOC-LF	6.70E-04
LPI DH3BX-XOC-LF	6.70E-04	LPI BW8AX-XOC-LF	6.70E-04
LPI BW8BX-XOC-LF	6.70E-04	LPI BW1X-XOC-LF	6.70E-04
RBIBS1BX-XOC-LF	6.70E-04	RBIBS1AX-XOC-LF	6.70E-04
RBIBW5BX-XOC-LF	6.70E-04	RBIBW5AX-XOC-LF	6.70E-04
SWS002CX-XOC-LF	6.70E-04	SWS002BX-XOC-LF	6.70E-04
SWS022BX-XOC-LF	6.70E-04	SWS024CX-XOC-LF	6.70E-04
SWS0218X-XOC-LF	6.70E-04	SWS0061X-XOC-LF	6.70E-04
SWS0064X-XOC-LF	6.70E-04	SWS038AX-XOC-LF	6.70E-04
SWS018CX-XOC-LF	6.70E-04	SWS037CX-XOC-LF	6.70E-04
SWS018AX-XOC-LF	6.70E-04	SWS037AX-XOC-LF	6.70E-04
SWS018BX-XOC-LF	6.70E-04	SWS0015X-XOC-LF	6.70E-04
SWS0014X-XOC-LF	6.70E-04	SWS037BX-XOC-LF	6.70E-04
SWS020AX-XOC-LF	6.70E-04	SWS022AX-XOC-LF	6.70E-04
SWS021AX-XOC-LF	6.70E-04	SWS0063X-XOC-LF	6.70E-04
SWS0086X-XOC-LF	6.70E-04	SWS038BX-XOC-LF	6.70E-04
SWS068X-XOC-LF	6.70E-04	SWS067X-XOC-LF	6.70E-04
SWS066X-XOC-LF	6.70E-04	HPRMU24X-XCC-LF	6.70E-04
HPRMU23X-XCC-LF	6.70E-04	HPRMU14X-XCC-LF	6.70E-04
HPRMU15X-XCC-LF	6.70E-04	HPRDH7BX-XCC-LF	6.70E-04
HPRDH7AX-XCC-LF	6.70E-04	EFW2800A-VOC-LF	1.10E-02
EFW2802B-VOC-LF	1.10E-02	EFW2617B-VOC-LF	1.10E-02
EFW2670A-VOC-LF	1.10E-02	EFW2627A-VOC-LF	1.10E-02
EFW2620B-VOC-LF	1.10E-02	EFW2626B-VOC-LF	1.10E-02
EFW2667A-VOC-LF	1.10E-02	EFW2666B-VOC-LF	1.10E-02
EFW00X3A-VOC-LF	1.10E-02	EFW00X4B-VOC-LF	1.10E-02
EFW00X1A-VOC-LF	1.10E-02	EFW00X2B-VOC-LF	1.10E-02
LPR1414A-VOC-LF	1.10E-02	LPR1415B-VOC-LF	1.10E-02
SWS3824-VOC-LF	1.10E-02	SWS3641B-VOC-LF	1.10E-02
SWS3644A-VOC-LF	1.10E-02	SWS3646A-VOC-LF	1.10E-02
SWS3645A-VOC-LF	1.10E-02	SWS3809A-VOC-LF	1.10E-02
EFW00Y1A-VCC-LF	1.90E-02	EFW00Y2B-VCC-LF	1.90E-02
SWS3806A-VCC-LF	1.90E-02	SWS3807B-VCC-LF	1.90E-02
LPI1407A-VCC-LF	1.90E-02	LPI1408B-VCC-LF	1.90E-02
HPI1220A-VCC-LF	1.90E-02	HPI1228B-VCC-LF	1.90E-02
HPI1227B-VCC-LF	1.90E-02	HPI1219A-VCC-LF	1.90E-02

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VARIABLE	VALUE
LPR1405A-VCC-LF	1.90E-02
LPI1401A-VCC-LF	1.90E-02
RB12400B-VCC-LF	1.90E-02
SWS3821-VCC-LF	1.90E-02
SWS3810B-VCC-LF	1.90E-02
SWS3803-VCC-LF	1.90E-02
SWS001CX-COC-LF	6.10E-04
EFW00006X-CCC-LF	6.70E-04
EFW0138X-CCC-LF	6.70E-04
EFW0002X-CCC-LF	6.70E-04
EFW0003X-CCC-LF	6.70E-04
EFW0108X-CCC-LF	6.70E-04
EFW0099X-CCC-LF	6.70E-04
EFW0262X-CCC-LF	6.70E-04
HPIU348X-CCC-LF	6.70E-04
HPI1212X-CCC-LF	6.70E-04
HPI1215X-CCC-LF	6.70E-04
HPI1214X-CCC-LF	6.70E-04
HPIU19CX-CCC-LF	6.70E-04
HPIU34DX-CCC-LF	6.70E-04
LPIDH2AX-CCC-LF	6.70E-04
LPIDH14B-CCC-LF	6.70E-04
LPIDH17-CCC-LF	6.70E-04
LPIDH18-CCC-LF	6.70E-04
LP1BW48X-CCC-LF	6.70E-04
RB1BS48X-CCC-LF	6.70E-04
RB1BW6AX-CCC-LF	6.70E-04
EFW00Y3X-NOC-LF	1.10E-02
LPI1428A-NOC-LF	1.10E-02
SWS3805-NCC-LF	1.90E-02
SWS3804-NCC-LF	1.90E-02
SWS3642B-V00-LF	1.40E-02
SWS3643A-V00-LF	1.40E-02
HPIU208X-XOC-UTM	1.10E-04
HPIU20AX-XOC-UTM	1.10E-04
HPI1224X-XOC-UTM	1.10E-04
HPIU45DX-XOC-UTM	1.10E-04
HPIU18CX-XOC-UTM	1.10E-04
HPIU45AX-XOC-UTM	1.10E-04
HPIU458X-XOC-UTM	1.10E-04
HPI1232X-XOC-UTM	1.10E-04
HPI1231X-XOC-UTM	1.10E-04
HPIMU25X-XOC-UTM	1.10E-04
HPIMU17X-XOC-UTM	1.10E-04
LPIDH3AX-XOC-UTM	1.10E-04
LP1BW88X-XOC-UTM	1.10E-04
RB1BS1BX-XOC-UTM	1.10E-04

VARIABLE	VALUE
PR1406B-VCC-LF	1.90E-02
LPI1400B-VCC-LF	1.90E-02
RBI2401A-VCC-LF	1.90E-02
SWS3802-VCC-LF	1.90E-02
SWS3808A-VCC-LF	1.90E-02
SWS3822-VCC-LF	1.90E-02
SWS001BX-COC-LF	6.10E-04
EFW013AX-CCC-LF	6.70E-04
EFW00Q1X-CCC-LF	6.70E-04
EFW00Q4X-CCC-LF	6.70E-04
EFW010AX-CCC-LF	6.70E-04
EFW0098X-CCC-LF	6.70E-04
EFW0261X-CCC-LF	6.70E-04
EFW00Q5X-CCC-LF	6.70E-04
HPIU34AX-CCC-LF	6.70E-04
HPIU34CX-CCC-LF	6.70E-04
HPI1213X-CCC-LF	6.70E-04
HPI1211X-CCC-LF	6.70E-04
HPIU19AX-CCC-LF	6.70E-04
HPIU19BX-CCC-LF	6.70E-04
LPIDH2BX-CCC-LF	6.70E-04
LPIDH14A-CCC-LF	6.70E-04
LPIDH13B-CCC-LF	6.70E-04
LPIDH13A-CCC-LF	6.70E-04
LP1BW4AX-CCC-LF	6.70E-04
RB1BS4AX-CCC-LF	6.70E-04
RB1BW6BX-CCC-LF	6.70E-04
EFW00Y4X-NOC-LF	1.10E-02
LPI1429B-NOC-LF	1.10E-02
SWS3B41-NCC-LF	1.90E-02
SWS3840-NCC-LF	1.90E-02
SWS3640B-V00-LF	1.40E-02
SWS3820A-V00-LF	1.40E-02
HPIU18BX-XOC-UTM	1.10E-04
HPIU18AX-XOC-UTM	1.10E-04
HPI1223X-XOC-UTM	1.10E-04
HPIU20CX-XOC-UTM	1.10E-04
HPIMU26X-XOC-UTM	1.10E-04
HPI1234X-XOC-UTM	1.10E-04
HPI1233X-XOC-UTM	1.10E-04
HPIU45CX-XOC-UTM	1.10E-04
HPIOBW2X-XOC-UTM	1.10E-04
HPIMU16X-XOC-UTM	1.10E-04
HPIOBW3X-XOC-UTM	1.10E-04
LPIDH3BX-XOC-UTM	1.10E-04
RB1BW5BX-XOC-UTM	1.10E-04
RB1BW5AX-XOC-UTM	1.10E-04

• • • • VALUE BLOCK TABLE • • • •
(ANOSMVL - LEVEL 3)

VARIABLE	VALUE	VARIABLE	VALUE
RBIBS1AX-XOC-UTM	1.10E-04	SWS002CX-XOC-UTM	1.10E-04
SWS002BX-XOC-UTM	1.10E-04	SWS022BX-XOC-UTM	1.10E-04
SWS024CX-XOC-UTM	1.10E-04	SWS0218X-XOC-UTM	1.10E-04
SWS0061X-XOC-UTM	1.10E-04	SWS0064X-XOC-UTM	1.10E-04
SWS038AX-XOC-UTM	1.10E-04	SWS018CX-XOC-UTM	1.10E-04
SWS037CX-XOC-UTM	1.10E-04	SWS018AX-XCC-UTM	1.10E-04
SWS037AX-XOC-UTM	1.10E-04	SWS018BX-XOC-UTM	1.10E-04
SWS0015X-XOC-UTM	1.10E-04	SWS0014X-XOC-UTM	1.10E-04
SWS037BX-XOC-UTM	1.10E-04	SWS020AX-XOC-UTM	1.10E-04
SWS022AX-XOC-UTM	1.10E-04	SWS021AX-XOC-UTM	1.10E-04
SWS0063X-XOC-UTM	1.10E-04	SWS0086X-XOC-UTM	1.10E-04
SWS038BX-XOC-UTM	1.10E-04	SWS068X-XOC-UTM	1.10E-04
SWS067X-XOC-UTM	1.10E-04	SWS066X-XOC-UTM	1.10E-04
LPIDH8AX-XCO-UTM	1.10E-04	LPIDH8BX-XCO-UTM	1.10E-04
RBIDH10X-XCO-UTM	1.10E-04	HPRMU24X-XCC-UTM	3.00E-05
HPRMU23X-XCC-UTM	3.00E-05	HPRMU14X-XCC-UTM	3.00E-05
HPRMU15X-XCC-UTM	3.00E-05	HPRDH7BX-XCC-UTM	3.00E-05
HPRDH7AX-XCC-UTM	3.00E-05	EFW00X2B-VOC-UTM	1.10E-04
EFW2670A-VOC-UTM	1.10E-04	EFW2627A-VOC-UTM	1.10E-04
EFW2620B-VOC-UTM	1.10E-04	EFW2626B-VOC-UTM	1.10E-04
EFW2802B-VOC-UTM	1.10E-04	EFW2667A-VOC-UTM	1.10E-04
EFW2666B-VOC-UTM	1.10E-04	EFW2617B-VOC-UTM	1.10E-04
EFW00X3A-VOC-UTM	1.10E-04	EFW00X4B-VOC-UTM	1.10E-04
EFW00X1A-VOC-UTM	1.10E-04	EFW2800A-VOC-UTM	1.10E-04
LPR1414A-VOC-UTM	1.10E-04	LPR1415B-VOC-UTM	1.10E-04
SWS3641B-VOC-UTM	1.10E-04	SWS3644A-VOC-UTM	1.10E-04
SWS3646A-VOC-UTM	1.10E-04	SWS3645A-VOC-UTM	1.10E-04
SWS3809A-VOC-UTM	1.10E-04	EFW00Y1A-VCC-UTM	3.00E-05
EFW00Y2B-VCC-UTM	3.00E-05	SWS3806A-VCC-UTM	3.00E-05
SWS3807B-VCC-UTM	3.00E-05	LPI1407A-VCC-UTM	3.00E-05
LPI1408B-VCC-UTM	3.00E-05	HPI1220A-VCC-UTM	3.00E-05
HPI1219A-VCC-UTM	3.00E-05	HPI1228B-VCC-UTM	3.00E-05
HPI1227B-VCC-UTM	3.00E-05	LPR1405A-VCC-UTM	3.00E-05
LPR1406B-VCC-UTM	3.00E-05	RBI2400B-VCC-UTM	3.00E-05
RBI2401A-VCC-UTM	3.00E-05	SWS3821-VCC-UTM	3.00E-05
SWS3802-VCC-UTM	3.00E-05	SWS3810B-VCC-UTM	3.00E-05
SWS3808A-VCC-UTM	3.00E-05	SWS3803-VCC-UTM	3.00E-05
SWS3822-VCC-UTM	3.00E-05	LPI1400B-VCC-UTM	3.00E-05
LPI1401A-VCC-UTM	3.00E-05	EFW00Y3X-NOC-UTM	1.10E-04
EFW00Y4X-NOC-UTM	1.10E-04	LPI1422A-NOC-UTM	1.10E-04
LPI1429B-NOC-UTM	1.10E-04	SWS3805-NCC-UTM	1.10E-04
SWS3841-NCC-UTM	1.10E-04	SWS3804-NCC-UTM	1.10E-04
SWS3840-NCC-UTM	1.10E-04	SWS3642B-VCC-UTM	3.00E-05
SWS3640B-VCC-UTM	3.00E-05	SWS3643A-VCC-UTM	3.00E-05
SWS3820A-VCC-UTM	3.00E-05	SRV-1-LF	1.00E-05
SRV-2-LF	1.00E-05	ERV-LF	3.00E-04
BLKVLV-LF	1.90E-02	BLKVLV-GATE	1.00E+00

• • • • VALUE BLOCK TABLE • • • •
(ANOSMVL - LEVEL 3)

VARIABLE	VALUE	VARIABLE	VALUE
EFWOP7BA-PMD-LF	1.00E-02	EFWOP7AX-PTD-LF	3.60E-02
HPIP36CB-PMD-LF	1.00E-02	HPIP36BA-PMD-LF	1.00E-02
HPIP36AA-PMD-LF	1.00E-02	LPIOP34A-PMD-LF	1.00E-02
LPIOP34B-PMD-LF	1.00E-02	RBIP35BB-PMD-LF	1.00E-02
RBIP35AA-PMD-LF	1.00E-02	SWSOP4CB-PMD-LF	1.60E-02
SWSOP4BA-PMD-LF	1.60E-02	SWSOP4CB-PAC-LF	1.60E-02
SWSOP4BA-PAC-LF	1.60E-02	EFWOP7BA-PMD-UTM	2.80E-04
HPIP36BA-PMD-UTM	2.80E-04	HPIP36AA-PMD-UTM	2.80E-04
HPIP36CB-PMD-UTM	2.80E-04	LPIOP34A-PMD-UTM	2.80E-04
LPIOP34B-PMD-UTM	2.80E-04	RBIP35BB-PMD-UTM	2.80E-04
RBIP35AA-PMD-UTM	2.80E-04	SWSOP4CB-PMD-UTM	0.
SWSOP4BA-PMD-UTM	0.	SWSOP4CB-PAC-UTM	0.
SWSOP4BA-PAC-UTM	0.	EFWOP7AX-PTD-UTM	6.80E-04
DG-CM	1.30E-02	BATT-CM	3.10E-02
EFWSP-CM	1.10E-03	LPSP-CM	1.00E-03
RBSIP-CM	1.00E-03	HPSP-CM	1.00E-03
SWSP-CM	6.80E-03	SRV-CM	1.30E-06
HPSV-CM	2.70E-03	EFWSV-CM	2.70E-03
RBC-SWL-CM	3.40E-03	HPISV-CM	3.00E-03
LPSV-CM	3.00E-03	LPRSV-CM	3.00E-03
RBSIV-CM	3.00E-03	SWSV-08-10-CM	3.00E-03
SWSV-40-41-CM	3.00E-03	SWSV-02-03-CM	3.00E-03
SWSV-21-22-CM	3.00E-03	SWSV-04-05-CM	3.00E-03
SWSV-06-07-CM	3.00E-03	SWSV-40-42-CM	3.00E-03
CSA-CM	4.70E-04	CSB-CM	4.70E-04
A1-4160VAC-LF	9.96E-01	A2-4160VAC-LF	9.96E-01
B5-480VAC-LF	9.96E-01	B6-480VAC-LF	9.96E-01
A3-4160VAC-LF	9.96E-01	A4-4160VAC-LF	9.96E-01
B55-480VAC-LF	9.96E-01	B56-480VAC-LF	9.96E-01
B51-480VAC-LF	9.96E-01	B52-480VAC-LF	9.96E-01
B61-480VAC-LF	9.96E-01	B62-480VAC-LF	9.96E-01
RA1-125VDC-LF	1.40E-03	RA2-125VDC-LF	1.40E-03
D11-125VDC-LF	1.40E-03	D21-125VDC-LF	1.40E-03
D02-125VDC-LF	1.40E-03	D01-125VDC-LF	1.40E-03
LF-DC-RS1	1.40E-03	LF-DC-RS2	1.40E-03
B55-480VAC-UTM	0.	B56-480VAC-UTM	0.
B5-480VAC-UTM	0.	A3-4160VAC-UTM	0.
B51-480VAC-UTM	0.	B52-480VAC-UTM	0.
A2-4160VAC-UTM	0.	B6-480VAC-UTM	0.
A4-4160VAC-UTM	0.	B61-480VAC-UTM	0.
B62-480VAC-UTM	0.	A1-4160VAC-UTM	0.
RA1-125VDC-UTM	0.	RA2-125VDC-UTM	0.
D11-125VDC-UTM	0.	D01-125VDC-UTM	0.
D21-125VDC-UTM	0.	D02-125VDC-UTM	0.
TM-DC-RS1	0.	TM-DC-RS2	0.
DG1-GEN-LF	5.50E-02	DG2-GEN-LF	5.50E-02
DG1-GEN-UTM	6.10E-03	DG2-GEN-UTM	6.10E-03

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• • • • VALUE BLOCK TABLE • • • •
(ANOSMVL - LEVEL 3)

VARIABLE	VALUE	VARIABLE	VALUE
LF-DC-Y11AC	2.00E-01	LF-DC-Y22AC	2.00E-01
LF-DC-Y11	2.00E-01	LF-DC-Y22	2.00E-01
LF-DC-Y11DC	2.00E-01	LF-DC-Y22DC	2.00E-01
LF-DC-RS1ALTAC	2.00E-01	LF-DC-RS2ALTAC	2.00E-01
BATTD07-ZBT-LF	9.80E-02	BATTD06-ZBT-LF	9.80E-02
BATTD07-ZBT-UTM	1.10E-03	BATTD06-ZBT-UTM	1.10E-03
D03-CHARGER-LF	3.60E-03	D04-CHARGER-LF	3.60E-03
LOSS-OSP	1.00E+00	TM-LOGIC-CM	2.00E-04
ACT-SG-SENS-CM	3.00E-05	ACT-EFAA-LF	1.20E-02
ACT-EFBB-LF	1.21E-02	ACT-EFXA-LF	1.20E-02
ACT-EFXB-LF	1.20E-02	LF-REF-RM95	6.20E-03
LF-REF-RM109	6.20E-03	LF-ECS-A1	1.00E-04
LF-ECS-B1	1.00E-04	LF-ECS-A2	1.00E-04
LF-ECS-B2	1.00E-04	LF-ECS-A4	1.00E-04
LF-ECS-B4	1.00E-04	LF-ECS-A5	3.80E-03
LF-ECS-B5	4.80E-03	LF-ECS-A6	4.80E-03
LF-ECS-B6	6.40E-03	DIG-LOG-CM-TM	1.70E-06
LF-ESF-T1	6.70E-03	LF-ESF-T2	6.70E-03
LF-ESF-T3	6.70E-03	LF-ESF-T4	6.70E-03
LF-ESF-T5	6.70E-03	LF-ESF-T6	6.70E-03
LF-ESF-T7	6.70E-03	LF-ESF-T8	6.70E-03
LF-ESF-TRS1	6.70E-03	LF-ESF-TRS2	6.70E-03
ESFA110X-LOG-LF	6.80E-03	ESFA210X-LOG-LF	6.80E-03
ESFA310X-LOG-LF	6.80E-03	ESFA116X-LOG-LF	6.80E-03
ESFA216X-LOG-LF	6.80E-03	ESFA316X-LOG-LF	6.80E-03
ESFA122-LOG-LF	6.80E-03	ESFA222-LOG-LF	6.80E-03
ESFA322-LOG-LF	6.80E-03	ESFA123-LOG-LF	6.80E-03
ESFA223-LOG-LF	6.80E-03	ESFA323-LOG-LF	6.80E-03
LF-ESF-TA01	1.90E-02	LF-ESF-TB01	1.90E-02
LF-ESF-TC01	1.90E-02	LF-ESF-TA120	1.90E-02
LF-ESF-TA220	1.90E-02	LF-ESF-TA320	1.90E-02
LF-ESF-TA119	1.90E-02	LF-ESF-TA219	1.90E-02
LF-ESF-TA319	1.90E-02	LF-ESF-TA04	7.20E-03
LF-ESF-TB04	7.20E-03	LF-ESF-TC04	7.20E-03
BISTABLEX08-CM	2.60E-03	BISTABLEX19-CM	2.60E-03
BISTABLEX20-CM	2.60E-03	ESF-TSL	0.
LF-SWS-VCH4A	4.50E-02	LF-SWS-VCH4B	4.50E-02
CST-TNK-LF	9.40E-03	BWST-TNK-LF	7.16E-01
HPRE35AX-HTX-LFB	4.90E-04	HPRE35BX-HTX-LFB	4.90E-04
HPRE35AX-HTX-UTM	5.00E-04	HPRE35BX-HTX-UTM	6.00E-04
LF-RBC-SEGC8	9.40E-01	LF-RBC-SEGAA	9.40E-01
LF-RBC-SEG8A	9.40E-01	LF-RBC-SEGDB	9.40E-01
LF-RBC-SWL1	4.90E-02	LF-RBC-SWL2	4.90E-02
FNB-MANACT	3.00E-03	OPF-OPEN-FNB	3.00E-03
HPRS-MANACT	3.00E-03	RBSR-MANACT	3.00E-03
LPRS-MANACT	3.00E-03	DG-MANACT	5.00E-02

* * * * VALUE BLOCK TABLE * * * *
(ANOSMVL - LEVEL 4)

VARIABLE	VALUE	VARIABLE	VALUE
EFW0019X-XOC-LF	1.20E-02	HPIU45AX-XOC-LF	1.20E-02
HPI1234X-XOC-LF	1.20E-02	HPIU45BX-XOC-LF	1.20E-02
HPI1233X-XOC-LF	1.20E-02	HPI1224X-XOC-LF	1.20E-02
HPI1223X-XOC-LF	1.20E-02	HPI1232X-XOC-LF	1.20E-02
HPIU45CX-XOC-LF	1.20E-02	HPI1231X-XOC-LF	1.20E-02
HPIU20CX-XOC-LF	1.20E-02	HPIU18CX-XOC-LF	1.20E-02
HPIOBW2X-XOC-LF	1.20E-02	HPIMU25X-XOC-LF	1.20E-02
HPIMU26X-XOC-LF	1.20E-02	HPIMU16X-XOC-LF	1.20E-02
HPIMU17X-XOC-LF	1.20E-02	HPIOBW3X-XOC-LF	1.20E-02
HPIU20AX-XOC-LF	1.20E-02	HPIU18AX-XOC-LF	1.20E-02
HPIU45DX-XOC-LF	1.20E-02	HPIU20BX-XOC-LF	1.20E-02
HPIU18BX-XOC-LF	1.20E-02	LPIDH3AX-XOC-LF	1.20E-02
LPIDH3BX-XOC-LF	1.20E-02	LPBW8AX-XOC-LF	1.20E-02
LPBW8BX-XOC-LF	1.20E-02	LPBW1X-XOC-LF	1.20E-02
RBIBS1BX-XOC-LF	1.20E-02	RBIBS1AX-XOC-LF	1.20E-02
RBIBW5BX-XOC-LF	1.20E-02	RBIBW5AX-XOC-LF	1.20E-02
SWS002CX-XOC-LF	1.20E-02	SWS002BX-XOC-LF	1.20E-02
SWS022BX-XOC-LF	1.20E-02	SWS024CX-XOC-LF	1.20E-02
SWS0218X-XOC-LF	1.20E-02	SWS0061X-XOC-LF	1.20E-02
SWS0064X-XOC-LF	1.20E-02	SWS038AX-XOC-LF	1.20E-02
SWS018CX-XOC-LF	1.20E-02	SWS037CX-XOC-LF	1.20E-02
SWS018AX-XOC-LF	1.20E-02	SWS037AX-XOC-LF	1.20E-02
SWS018BX-XOC-LF	1.20E-02	SWS0015X-XOC-LF	1.20E-02
SWS0014X-XOC-LF	1.20E-02	SWS037BX-XOC-LF	1.20E-02
SWS020AX-XOC-LF	1.20E-02	SWS022AX-XOC-LF	1.20E-02
SWS021AX-XOC-LF	1.20E-02	SWS0063X-XOC-LF	1.20E-02
SWS0086X-XOC-LF	1.20E-02	SWS038BX-XOC-LF	1.20E-02
SWS068X-XOC-LF	1.20E-02	SWS067X-XOC-LF	1.20E-02
SWS066X-XOC-LF	1.20E-02	HPRMU24X-XCC-LF	1.20E-02
HPRMU23X-XCC-LF	1.20E-02	HPRMU14X-XCC-LF	1.20E-02
HPRMU15X-XCC-LF	1.20E-02	HPRDH7BX-XCC-LF	1.20E-02
HPRDH7AX-XCC-LF	1.20E-02	EFW2800A-VOC-LF	8.80E-02
EFW2802B-VOC-LF	8.80E-02	EFW2617B-VOC-LF	8.80E-02
EFW2670A-VOC-LF	8.80E-02	EFW2627A-VOC-LF	8.80E-02
EFW2620B-VOC-LF	8.80E-02	EFW2626B-VOC-LF	8.80E-02
EFW2667A-VOC-LF	8.80E-02	EFW2666B-VOC-LF	8.80E-02
EFW00X3A-VOC-LF	8.80E-02	EFW00X4B-VOC-LF	8.80E-02
EFW00X1A-VOC-LF	8.80E-02	EFW00X2B-VOC-LF	8.80E-02
LPR1414A-VOC-LF	8.80E-02	LPR1415B-VOC-LF	8.80E-02
SWS3024-VOC-LF	8.80E-02	SWS3641B-VOC-LF	8.80E-02
SWS3644A-VOC-LF	8.80E-02	SWS3646A-VOC-LF	8.80E-02
SWS3645A-VOC-LF	8.80E-02	SWS3809A-VOC-LF	8.80E-02
EFW00Y1A-VCC-LF	9.60E-02	EFW00Y2B-VCC-LF	9.60E-02
SWS3806A-VCC-LF	9.60E-02	SWS3807B-VCC-LF	9.60E-02
LPI1407A-VCC-LF	9.60E-02	LPI1408B-VCC-LF	9.60E-02
HPI1220A-VCC-LF	9.60E-02	HPI1228B-VCC-LF	9.60E-02
HPI1227B-VCC-LF	9.60E-02	HPI1219A-VCC-LF	9.60E-02

• • • • VALUE BLOCK TABLE • • • •
(ANOSMVL - LEVEL 4)

VARIABLE	VALUE	VARIABLE	VALUE
LPR1405A-VCC-LF	9.60E-02	LPR1406B-VCC-LF	9.60E-02
LPI1401A-VCC-LF	9.60E-02	LPI1400B-VCC-LF	9.60E-02
RBI2400B-VCC-LF	9.60E-02	RBI2401A-VCC-LF	9.60E-02
SWS3821-VCC-LF	9.60E-02	SWS3802-VCC-LF	9.60E-02
SWS3810B-VCC-LF	9.60E-02	SWS3808A-VCC-LF	9.60E-02
SWS3803-VCC-LF	9.60E-02	SWS3822-VCC-LF	9.60E-02
SWS001CX-COC-LF	1.20E-02	SWS001BX-COC-LF	1.20E-02
EFW0006X-CCC-LF	1.20E-02	EFW013AX-CCC-LF	1.20E-02
EFW013BX-CCC-LF	1.20E-02	EFW00Q1X-CCC-LF	1.20E-02
EFW00Q2X-CCC-LF	1.20E-02	EFW00Q4X-CCC-LF	1.20E-02
EFW00Q3X-CCC-LF	1.20E-02	EFW010AX-CCC-LF	1.20E-02
EFW010BX-CCC-LF	1.20E-02	EFW0098X-CCC-LF	1.20E-02
EFW0099X-CCC-LF	1.20E-02	EFW0261X-CCC-LF	1.20E-02
EFW0262X-CCC-LF	1.20E-02	EFW00Q5X-CCC-LF	1.20E-02
HPIU34BX-CCC-LF	1.20E-02	HPIU34AX-CCC-LF	1.20E-02
HPI1212X-CCC-LF	1.20E-02	HPIU34CX-CCC-LF	1.20E-02
HPI1215X-CCC-LF	1.20E-02	HPI1213X-CCC-LF	1.20E-02
HPI1214X-CCC-LF	1.20E-02	HPI1211X-CCC-LF	1.20E-02
HPIU19CX-CCC-LF	1.20E-02	HPIU19AX-CCC-LF	1.20E-02
HPIU34DX-CCC-LF	1.20E-02	HPIU19BX-CCC-LF	1.20E-02
LPIDH2AX-CCC-LF	1.20E-02	LPIDH2BX-CCC-LF	1.20E-02
LPIDH14B-CCC-LF	1.20E-02	LPIDH14A-CCC-LF	1.20E-02
LPIDH17-CCC-LF	1.20E-02	LPIDH13B-CCC-LF	1.20E-02
LPIDH18-CCC-LF	1.20E-02	LPIDH13A-CCC-LF	1.20E-02
LPIBW4BX-CCC-LF	1.20E-02	LPIBW4AX-CCC-LF	1.20E-02
RBIBS4BX-CCC-LF	1.20E-02	RBIBS4AX-CCC-LF	1.20E-02
RBIBW6AX-CCC-LF	1.20E-02	RBIBW6BX-CCC-LF	1.20E-02
EFW00Y3X-NOC-LF	8.80E-02	EFW00Y4X-NOC-LF	8.80E-02
LPI1428A-NOC-LF	8.80E-02	LPI1429B-NOC-LF	8.80E-02
SWS3805-NCC-LF	9.60E-02	SWS3841-NCC-LF	9.60E-02
SWS3804-NCC-LF	9.60E-02	SWS3840-NCC-LF	9.60E-02
SWS3642B-V00-LF	9.10E-02	SWS3640B-V00-LF	9.10E-02
SWS3643A-V00-LF	9.10E-02	SWS3820A-V00-LF	9.10E-02
HPIU20BX-XOC-UTM	1.10E-04	HPIU18BX-XOC-UTM	1.10E-04
HPIU20AX-XOC-UTM	1.10E-04	HPIU18AX-XOC-UTM	1.10E-04
HPI1224X-XOC-UTM	1.10E-04	HPI1223X-XOC-UTM	1.10E-04
HPIU45DX-XOC-UTM	1.10E-04	HPIU20CX-XOC-UTM	1.10E-04
HPIU18CX-XOC-UTM	1.10E-04	HPIU26X-XOC-UTM	1.10E-04
HPIU45AX-XOC-UTM	1.10E-04	HPI1234X-XOC-UTM	1.10E-04
HPIU45BX-XOC-UTM	1.10E-04	HPI1233X-XOC-UTM	1.10E-04
HPI1232X-XOC-UTM	1.10E-04	HPIU45CX-XOC-UTM	1.10E-04
HPI1231X-XOC-UTM	1.10E-04	HPIOBW2X-XOC-UTM	1.10E-04
HPIU25X-XOC-UTM	1.10E-04	HPIU16X-XOC-UTM	1.10E-04
HPIU17X-XOC-UTM	1.10E-04	HPIOBW3X-XOC-UTM	1.10E-04
LPIDH3AX-XOC-UTM	1.10E-04	LPIDH3BX-XOC-UTM	1.10E-04
LPIBW8BX-XOC-UTM	1.10E-04	RBIBW5BX-XOC-UTM	1.10E-04
RBIBS1BX-XOC-UTM	1.10E-04	RBIBW5AX-XOC-UTM	1.10E-04

* * * * VALUE BLOCK TABLE * * * *
(ANOSMVL - LEVEL 4)

VARIABLE	VALUE	VARIABLE	VALUE
RBIBS1AX-XOC-UTM	1.10E-04	SWS002CX-XOC-UTM	1.10E-04
SWS002BX-XOC-UTM	1.10E-04	SWS022BX-XOC-UTM	1.10E-04
SWS024CX-XOC-UTM	1.10E-04	SWS021BX-XOC-UTM	1.10E-04
SWS0061X-XOC-UTM	1.10E-04	SWS0064X-XOC-UTM	1.10E-04
SWS038AX-XOC-UTM	1.10E-04	SWS018CX-XOC-UTM	1.10E-04
SWS037CX-XOC-UTM	1.10E-04	SWS018AX-XOC-UTM	1.10E-04
SWS037AX-XOC-UTM	1.10E-04	SWS018BX-XOC-UTM	1.10E-04
SWS0015X-XOC-UTM	1.10E-04	SWS0014X-XOC-UTM	1.10E-04
SWS037BX-XOC-UTM	1.10E-04	SWS020AX-XOC-UTM	1.10E-04
SWS022AX-XOC-UTM	1.10E-04	SWS021AX-XOC-UTM	1.10E-04
SWS0063X-XOC-UTM	1.10E-04	SWS0086X-XOC-UTM	1.10E-04
SWS038BX-XOC-UTM	1.10E-04	SWS068X-XOC-UTM	1.10E-04
SWS067X-XOC-UTM	1.10E-04	SWS066X-XOC-UTM	1.10E-04
LPIDH8AX-XCO-UTM	1.10E-04	LPIDH8BX-XCO-UTM	1.10E-04
RBIDH10X-XCO-UTM	1.10E-04	HPRMU24X-XCC-UTM	3.00E-05
HPRMU23X-XCC-UTM	3.00E-05	HPRMU14X-XCC-UTM	3.00E-05
HPRMU15X-XCC-UTM	3.00E-05	HPRDH7BX-XCC-UTM	3.00E-05
HPRDH7AX-XCC-UTM	3.00E-05	EFW00X2B-VOC-UTM	1.10E-04
EFW2670A-VOC-UTM	1.10E-04	EFW2627A-VOC-UTM	1.10E-04
EFW2620B-VOC-UTM	1.10E-04	EFW2626B-VOC-UTM	1.10E-04
EFW2802B-VOC-UTM	1.10E-04	EFW2667A-VOC-UTM	1.10E-04
EFW2666B-VOC-UTM	1.10E-04	EFW2617B-VOC-UTM	1.10E-04
EFW00X3A-VOC-UTM	1.10E-04	EFW00X4B-VOC-UTM	1.10E-04
EFW00X1A-VOC-UTM	1.10E-04	EFW2500A-VOC-UTM	1.10E-04
LPR1414A-VOC-UTM	1.10E-04	LPR1415B-VOC-UTM	1.10E-04
SWS3641B-VOC-UTM	1.10E-04	SWS3644A-VOC-UTM	1.10E-04
SWS3646A-VOC-UTM	1.10E-04	SWS3645A-VOC-UTM	1.10E-04
SWS3809A-VOC-UTM	1.10E-04	EFW00Y1A-VCC-UTM	3.00E-05
EFW00Y2B-VCC-UTM	3.00E-05	SWS3806A-VCC-UTM	3.00E-05
SWS3807B-VCC-UTM	3.00E-05	LPI1407A-VCC-UTM	3.00E-05
LPI1408B-VCC-UTM	3.00E-05	HPI1220A-VCC-UTM	3.00E-05
HPI1219A-VCC-UTM	3.00E-05	HPI1228B-VCC-UTM	3.00E-05
HPI1227B-VCC-UTM	3.00E-05	LPR1405A-VCC-UTM	3.00E-05
LPR1406B-VCC-UTM	3.00E-05	RBI2400B-VCC-UTM	3.00E-05
RBI2401A-VCC-UTM	3.00E-05	SWS3821-VCC-UTM	3.00E-05
SWS3802-VCC-UTM	3.00E-05	SWS3810B-VCC-UTM	3.00E-05
SWS3808A-VCC-UTM	3.00E-05	SWS3803-VCC-UTM	3.00E-05
SWS3822-VCC-UTM	3.00E-05	LPI1400B-VCC-UTM	3.00E-05
LPI1401A-VCC-UTM	3.00E-05	EFW00Y3X-NOC-UTM	1.10E-04
EFW00Y4X-NOC-UTM	1.10E-04	LPI1428A-NOC-UTM	1.10E-04
LPI1429B-NOC-UTM	1.10E-04	SWS3805-NCC-UTM	1.10E-04
SWS3841-NCC-UTM	1.10E-04	SWS3804-NCC-UTM	1.10E-04
SWS3840-NCC-UTM	1.10E-04	SWS3642B-VOC-UTM	3.00E-05
SWS3640B-VOC-UTM	3.00E-05	SWS3643A-VOC-UTM	3.00E-05
SWS3820A-VOC-UTM	3.00E-05	SRV-1-LF	1.00E-05
SRV-2-LF	1.00E-05	ERV-LF	3.00E-04
BLKVLV-LF	9.60E-02	BLKVLV-GATE	1.00E+00

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• • • VALUE BLOCK TABLE • • •
(ANOSMVL - LEVEL 4)

VARIABLE	VALUE	VARIABLE	VALUE
EFWOP7BA-PMD-LF	2.30E-02	EFWOP7AX-PTD-LF	4.90E-02
HPIIP36CB-PMD-LF	2.30E-02	HPIIP36BA-PMD-LF	2.30E-02
HPIIP36AA-PMD-LF	2.30E-02	LPIOP34A-PMD-LF	2.30E-02
LPIOP34B-PMD-LF	2.30E-02	RBIP35BB-PMD-LF	2.30E-02
RBIP35AA-PMD-LF	2.30E-02	SWSOP4CB-PMD-LF	5.90E-02
SWSOP4BA-PMD-LF	5.90E-02	SWSOP4CB-PAC-LF	5.90E-02
SWSOP4BA-PAC-LF	5.90E-02	EFWOP7BA-PMD-UTM	2.80E-04
HPIIP36BA-PMD-UTM	2.80E-04	HPIIP36AA-PMD-UTM	2.80E-04
HPIIP36CB-PMD-UTM	2.80E-04	LPIOP34A-PMD-UTM	2.80E-04
LPIOP34B-PMD-UTM	2.80E-04	RBIP35BB-PMD-UTM	2.80E-04
RBIP35AA-PMD-UTM	2.80E-04	SWSOP4CB-PMD-UTM	0.
SWSOP4BA-PMD-UTM	0.	SWSOP4CB-PAC-UTM	0.
SWSOP4BA-PAC-UTM	0.	EFWOP7AX-PTD-UTM	6.80E-04
DG-CM	6.60E-02	BATT-CM	1.60E-01
EFWSP-CM	3.60E-03	LPSP-CM	3.50E-03
RBSIP-CM	3.50E-03	HPSP-CM	3.50E-03
SWSP-CM	1.40E-02	SRV-CM	1.30E-06
HPSV-CM	3.00E-02	EFWSV-CM	3.00E-02
RBC-SWL-CM	3.00E-02	HPISV-CM	3.00E-02
LPSV-CM	3.00E-02	LPRSV-CM	3.00E-02
RBSIV-CM	3.00E-02	SWSV-08-10-CM	3.00E-02
SWSV-40-41-CM	3.00E-02	SWSV-02-03-CM	3.00E-02
SWSV-21-22-CM	3.00E-02	SWSV-04-05-CM	3.00E-02
SWSV-06-07-CM	3.00E-02	SWSV-40-42-CM	3.00E-02
CSA-CM	4.70E-04	CSB-CM	4.70E-04
A1-4160VAC-LF	9.99E-01	A2-4160VAC-LF	9.99E-01
B5-480VAC-LF	9.99E-01	B6-480VAC-LF	9.99E-01
A3-4160VAC-LF	9.99E-01	A4-4160VAC-LF	9.99E-01
B55-480VAC-LF	9.99E-01	B56-480VAC-LF	9.99E-01
B51-480VAC-LF	9.99E-01	B52-480VAC-LF	9.99E-01
B61-480VAC-LF	9.99E-01	B62-480VAC-LF	9.99E-01
RA1-125VDC-LF	1.30E-02	RA2-125VDC-LF	1.30E-02
D11-125VDC-LF	1.30E-02	D21-125VDC-LF	1.30E-02
D02-125VDC-LF	1.30E-02	D01-125VDC-LF	1.30E-02
LF-DC-RS1	1.30E-02	LF-DC-RS2	1.30E-02
B55-480VAC-UTM	0.	B56-480VAC-UTM	0.
B5-480VAC-UTM	0.	A3-4160VAC-UTM	0.
B51-480VAC-UTM	0.	B52-480VAC-UTM	0.
A2-4160VAC-UTM	0.	B6-480VAC-UTM	0.
A4-4160VAC-UTM	0.	B61-480VAC-UTM	0.
B62-480VAC-UTM	0.	A1-4160VAC-UTM	0.
RA1-125VDC-UTM	0.	RA2-125VDC-UTM	0.
D11-125VDC-UTM	0.	D01-125VDC-UTM	0.
D21-125VDC-UTM	0.	D02-125VDC-UTM	0.
TM-DC-RS1	0.	TM-DC-RS2	0.
DG1-GEN-LF	1.60E-01	DG2-GEN-LF	1.60E-01
DG1-GEN-UTM	6.10E-03	DG2-GEN-UTM	6.10E-03

• • • • VALUE BLOCK TABLE • • • •
(ANOSMVL - LEVEL 4)

VARIABLE	VALUE	VARIABLE	VALUE
LF-DC-Y11AC	4.80E-01	LF-DC-Y22AC	4.80E-01
LF-DC-Y11	4.80E-01	LF-DC-Y22	4.80E-01
LF-DC-Y11DC	4.80E-01	LF-DC-Y22DC	4.80E-01
LF-DC-RS1ALTAC	4.80E-01	LF-DC-RS2ALTAC	4.80E-01
BATTD07-ZBT-LF	2.90E-01	BATTD06-ZBT-LF	2.90E-01
BATTD07-ZBT-UTM	1.10E-03	BATTD06-ZBT-UTM	1.10E-03
D03-CHARGER-LF	1.90E-02	D04-CHARGER-LF	1.90E-02
LOSS-OSP	1.00E+00	TM-LOGIC-CM	2.00E-04
ACT-SG-SENS-CM	3.00E-05	ACT-EFAA-LF	3.70E-02
ACT-EFBB-LF	3.70E-02	ACT-EFXA-LF	3.70E-02
ACT-EFXB-LF	3.70E-02	LF-REF-RM95	1.30E-02
LF-REF-RM109	1.30E-02	LF-ECS-A1	1.00E-04
LF-ECS-B1	1.00E-04	LF-ECS-A2	1.00E-04
LF-ECS-B2	1.00E-04	LF-ECS-A4	1.00E-04
LF-ECS-B4	1.00E-04	LF-ECS-A5	3.80E-03
LF-ECS-B5	4.80E-03	LF-ECS-A6	4.80E-03
LF-ECS-B6	6.40E-03	DIG-LOG-CM-TM	1.70E-06
LF-ESF-T1	3.20E-02	LF-ESF-T2	3.20E-02
LF-ESF-T3	3.20E-02	LF-ESF-T4	3.20E-02
LF-ESF-T5	3.20E-02	LF-ESF-T6	3.20E-02
LF-ESF-T7	3.20E-02	LF-ESF-T8	3.20E-02
LF-ESF-TRS1	3.20E-02	LF-ESF-TRS2	3.20E-02
ESFA110X-LOG-LF	3.20E-02	ESFA210X-LOG-LF	3.20E-02
ESFA310X-LOG-LF	3.20E-02	ESFA116X-LOG-LF	3.20E-02
ESFA216X-LOG-LF	3.20E-02	ESFA316X-LOG-LF	3.20E-02
ESFA122-LOG-LF	3.20E-02	ESFA222-LOG-LF	3.20E-02
ESFA322-LOG-LF	3.20E-02	ESFA123-LOG-LF	3.20E-02
ESFA223-LOG-LF	3.20E-02	ESFA323-LOG-LF	3.20E-02
LF-ESF-TA01	4.40E-02	LF-ESF-TB01	4.40E-02
LF-ESF-TC01	4.40E-02	LF-ESF-TA120	4.40E-02
LF-ESF-TA220	4.40E-02	LF-ESF-TA320	4.40E-02
LF-ESF-TA119	4.40E-02	LF-ESF-TA219	4.40E-02
LF-ESF-TA319	4.40E-02	LF-ESF-TA04	3.20E-02
LF-ESF-TB04	3.20E-02	LF-ESF-TC04	3.20E-02
BISTABLEX08-CM	9.20E-03	BISTABLEX19-CM	9.20E-03
BISTABLEX20-CM	9.20E-03	ESF-TSL	0.
LF-SWS-VCH4A	1.80E-01	LF-SWS-VCH4B	1.80E-01
CST-TNK-LF	9.51E-02	BWST-TNK-LF	9.463E-01
HPRE35AX-HTX-LFB	4.90E-03	HPRE35BX-HTX-LFB	4.90E-03
HPRE35AX-HTX-UTM	6.00E-04	HPRE35BX-HTX-UTM	6.00E-04
LF-RBC-SEGCB	1.00E+00	LF-RBC-SEGAA	1.00E+00
LF-RBC-SEGBA	1.00E+00	LF-RBC-SEGDB	1.00E+00
LF-RBC-SWL1	1.80E-01	LF-RBC-SWL2	1.80E-01
FNB-MANACT	3.00E-03	OPF-OPEN-FNB	3.00E-03
HPRS-MANACT	3.00E-03	RBSR-MANACT	3.00E-03
LPRS-MANACT	3.00E-03	DG-MANACT	5.00E-02

Appendix D

FIRE RISK ANALYSIS FOR
ARKANSAS NUCLEAR ONE-1 NUCLEAR POWER PLANT

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APPENDIX D FIRE RISK ANALYSIS FOR ARKANSAS NUCLEAR ONE-1 NUCLEAR POWER PLANT

1. ANALYSIS METHODS AND ASSUMPTIONS

1.1 Introduction

Based on plant operating experience over the last 20 years, it has been observed that nuclear power plants will have three to four significant fires over their operating lifetimes (i.e., on the order of one fire every ten years or a frequency of $.1/R_x$ yr). Previous probabilistic risk assessments (PRAs) have shown that fires are a significant contributor to the overall core melt probability, contributing anywhere from 7 percent to 50 percent of the total (considering contributions from internal, seismic, flood, fire and other events). Because of the relatively high core melt contribution, fires need to be examined in more detail.

Large strides have been made in fire protection for nuclear power plants in recent years. In particular, 10 CFR 50, Appendix R and Branch Technical Positions 9.5-1 have led to improved fire protection measures at nuclear power plants. However, even with these improvements, there is still a residual risk from fires. These residual risks have not yet been quantified with the exception of those plants that have performed fire related PRAs. The purpose of this section is to quantify the risks due to fires at the Arkansas Nuclear One-1 nuclear power plant.

1.2 Scope

- a. This analysis will evaluate for the Arkansas Nuclear One-1 nuclear power plant the residual risk from fires that could affect decay heat removal systems. The results of this analysis, the quantified risk and any modifications that may be proposed, will be compared to risks from other initiating events with the modifications evaluated for risk reduction and cost. This "value impact" analysis will then ultimately determine the optimum set of modifications to reach the desired reduction in risk.
- b. The analysis is not intended to be a comprehensive fire PRA. An analysis of this sort would be beyond the scope of TAP A-45. The intent of this analysis is to quantify significant fire threats to a plant. In doing so, existing procedures used in published fire PRAs will be employed, but simplifying assumptions will be made to allow the analysis to proceed without requiring the level of detail required for these other fire PRAs. In that respect, this analysis embodies the basic approaches used by previous fire PRAs but is by necessity a scaled down version of these PRAs.

- c. Wherever possible, data from previous fire PRAs will be used. Since this analysis embodies what has been done in previous PRAs, use of this data will provide consistency in how the data is used and allow some comparison between the analyses. The data is based on that found in the Limerick [1], Millstone 3 [2] and Seabrook [3] fire PRAs (three of the most recent fire PRAs).
- d. Because this analysis is a scaled down version of a fire PRA, quantification of the risk due to a fire will be presented as a point estimate. Uncertainty values will not be formally determined. However, insights into the potential major areas of uncertainty will be identified.
- e. The systems to be examined and quantified are based on the event and fault trees developed by the systems analysts, and thus are constrained by the level of detail of the systems analysis effort. In a similar fashion, readily available plant documentation determines the level of physical detail used in this analysis. Documentation obtained for this effort is treated as accurate and up to date unless otherwise identified. No attempt is made to physically verify the locations of selected cables referenced in this documentation.
- f. A screening cutoff of $1.0E-06$ for fire-induced core melt probability was used to determine if further quantification efforts in an area were warranted. This was based on internal events analyses that showed core melt probabilities from internal events alone was on the order of $1.0E-04$ so that fires less than $1.0E-06$ would be insignificant.

1.3 Analysis Methodology

The approach used in determining the risk from a fire at the Arkansas Nuclear One-1 Nuclear Power Plant used four basic steps:

- a. Initial screening for potentially important fire areas
- b. Plant visit for verification
- c. Quantification
- d. Proposed modifications to reduce the fire risk in affected areas

1.3.1 Initial Screening for Potentially Important Fire Areas

The purpose of this step was to determine which areas in the plant had "enough" front line or support systems in their boundaries such that fire damage could invalidate all the success branches on the transient event tree. At this stage of the screening, the capability to remotely shut down the plant via local or remote stations was not considered.

The sequence of steps taken during this first screening were as follows:

- a. Internal event analysis data was obtained as the starting point. The specific data included event trees, fault trees, and simplified piping diagrams used in constructing the fault trees.
- b. A transient event tree was examined to determine those front line systems or combinations of systems that would lead to an "OK" state on the event tree. Also obtained from the event tree were the success criteria of front-line systems.
- c. The next step was to identify the support systems required for the front line systems identified above. Support systems included cooling systems and AC/DC power requirements. The success criteria for the support systems were obtained from the systems analysis. For all systems except the power conversion system, the fault trees for the front-line systems were used to determine the support systems. For the power conversion system, plant schematics and piping diagrams were used to determine required support systems.
- d. Once the systems of interest were identified, all available documentation (principally 10 CFR 50, Appendix R submittals, FSARs, and plant supplied information and drawings) was analyzed to determine those areas where "enough" components could be disabled by a fire. Based on this screening, a small set of areas were left for further analysis.

To this point in the analysis, everything was based on a review of available documentation.

1.3.2 Plant Visit

A visit was made to the plant with the following objectives:

- a. Based on the initial screening, the general location of cables and components of the systems of interest were known. The next step was to visit the plant and see the physical arrangements in each of these areas. The plant visit also provided a means to determine locations of major fuel sources and some estimate of the x,y,z coordinates of components and cables. This location information was necessary for the quantification step.
- b. The second purpose of the plant visit was to confirm with plant personnel that the documentation that was being used was in fact the best available information and to get clarification about any questions that might have arisen in a review of the documentation.

1.3.3 Quantification

For those areas remaining after the initial screening, the next step was to run a deterministic fire code (COMPBRN) [4] to ascertain if an exposure fire could damage the components/cables of interest in the area. The types of exposure fires utilized were a trash can fire and an acetone spill of 10 gallons. Based on the code results, if damage occurred then the analysis continued. If no damage occurred, the analysis was complete. The code calculations not only included cable failure from ignition, but also failure prior to ignition resulting from melting of the internal conductors. The melting is a function of temperature and time at temperature. Cable damage temperatures are discussed in the assumptions section.

Other factors were also considered at this stage to determine if further quantification of fire risk in a particular area should continue. Because control rooms are continually manned, the chance that a fire would not be suppressed prior to causing damage to enough of the systems of interest is less than for other, nonoccupied areas with the same basic fuel load. However, if "enough" of the systems of interest were bunched together, i.e., on a single control board section, a fire could potentially disable these safety systems. If this arrangement occurred, the control room fire risk was quantified; otherwise, it was not. Additionally, cables in any area that had a thermal blanket (like KAOWOOL) were assumed to survive a fire equal to the blanket manufacturer's rating.

An event tree was utilized to estimate the risk of core melt from a fire in those areas identified in the initial screening and not eliminated by code calculations. The basic elements of the event tree are as follows:

- a. Fire occurrence frequency for the area of interest.
- b. Suppression system probability (both for automatic and manual suppression activity).
- c. Operator error and random failures of unaffected systems from a remote shutdown location.

Each of these elements is discussed below:

- a. The fire frequency for each area was based on historical fire occurrence data found in previous fire PRAs. Table D.1 summarizes this data. Several specific rooms in nuclear power plants have their own fire frequency values, while other areas are only included in a building frequency. Data for specific rooms was used "as is" in this analysis. For rooms not included in the listing, a weighting factor was used to reduce a building frequency to a room frequency. Depending on the available information, either a cable weighting or an area weighting approach was used.

For Arkansas Nuclear One-1, a cable weighting approach is used. The cable weighting is a ratio of the the weight of cables in the room of interest to the weight of cables found in the rest of the building. (The area weighting method is a ratio of the individual room floor area to that of the building it is in.) Both of these methods have been used in previous fire PRAs.

The presence of an automatic fire detection system (such as a smoke detector) is not evaluated since in almost all cases plants have installed detectors throughout the plant. A probability is not assigned for detector failure because the detectors are highly reliable (on the order of 95 to 99 percent) and the probability of manual suppression incorporates detector response time. This step is consistent with prior fire PRAs.

- b. Both automatic (i.e., Halon, CO₂, water) and manual suppression systems were considered. Automatic suppression systems were assumed to actuate during the fire and immediately extinguish the fire. The reliabilities for the various automatic systems are given in Table D.2 and represent the probability that an automatic system malfunctions (i.e. does not actuate). Manual suppression probability is based on the time that the computer code predicts for the fire to damage the components of interest. Table D.3 lists the nonsuppression probabilities. (Note that these probabilities do not include suppression damage, but rather just the probability that the fire can be put out in a given time frame.)
- c. Any unaffected systems that can be either locally or remotely used to provide decay heat removal were evaluated. For fires that required the use of a remote shutdown location, the chance of incorrect operation was evaluated. This human error probability was taken to be .1, based on Seabrook [3] PRA data for the conditional frequency that the operators will not do the right thing at auxiliary shutdown panels. Next, the reliability of the remaining decay heat removal system(s) was evaluated using the fault trees for the plant. Values for the random failure of a single train are on the order of 1.0E-02/demand.

As an additional step, which should be treated only as a bounding calculation, failure of systems due to manual suppression damage was evaluated. Any components not affected by the fire were assumed to fail when manual suppression occurs. For this situation, the core melt probabilities were requantified. Damage from automatic suppression systems in the room or inadvertent operation in other locations was not assessed.

(It is recognized that this is a valid concern, and that automatic suppression damage potentially could increase the overall core melt probability, however the data bases and experimental data required to incorporate this concern do not presently exist and the generation of this type of data is clearly beyond the scope of the TAP A-45 analysis.)

1.3.4 Proposed Modifications

Based on the quantification steps, engineering modifications were proposed for those areas which have a fire-induced core melt probability contribution of greater than $1.0E-06/Rx$ yr. The proposed modifications were of the following nature:

- a. Installation of automatic suppression.
- b. Cable wrapping (only applicable if the cables being wrapped would not overheat from internal heat generation).
- c. Construction of barriers to isolate portions of an area.
- d. Automation of manual systems to improve their reliability.
- e. Rerouting of affected cables or physically moving redundant components to other locations.
- f. Procedural changes

For any proposed modification, a core melt probability was recalculated to indicate how much the core melt probability could be reduced. The quantification was performed by modifying the event tree to reflect the modification.

The proposed modifications were supplied to an architectural firm for cost evaluation during a second plant visit. A "value impact" analysis considering the core melt contributions from fires and other external and internal events was used to determine the "best modification" in an overall sense.

1.4 Assumptions

Most of the assumptions are based on insights gained from previous fire PRAs or from the historical fire data base. The assumptions are as follows:

- a. Only fires in a single area (room) will be considered, i.e., no room to room issues will be addressed.

From a review of previous PRAs, only single areas (rooms) have been significant contributors to the core melt probability from a fire. Although this procedure neglects certain equipment vulnerabilities (smoke and suppression system actuation outside the fire area) and the effects of ventilation system operation, these factors are assumed to be of less risk significance than single area fires.

Because no room to room issues are being considered, barriers between adjacent areas are assumed to remain intact during a fire, limiting the fire to only one area. Thus, the effectiveness of an area's fire barrier, either from a deterministic or probabilistic standpoint, will not be evaluated. It is assumed that the fire barriers, including walls, doors, dampers, and penetrations, have been properly constructed to ensure their integrity in a fire.

- b. Fire frequency and suppression reliability is based on available industry data.

Existing data was used to the extent possible in determining the threat of fire to a plant. Fire frequency data has been collected for all the fire PRAs performed. This data has been typically used as generic data that is modified to reflect the actual fire experiences at the plant. For this analysis, the fire frequency used was the mean value of the generic data; the plant's fire history was not used to modify this generic data. Suppression system data was based on nuclear insurers data for installed automatic systems [2] and on data from work done for the HTGR fire methodology PRA.[5] This data has been used by almost all previous fire PRAs when evaluating manual suppression probabilities.

- c. Two transient combustible exposure fires were assumed to bound all transient and electrically initiated fires.

The following exposure fires were be used in the computer calculations:

1. Trashcan size container of refuse with an energy output of approximately 500 kW for 15 minutes. The total heat released is approximately $4.5E+08$ Joules. For transient combustibles, such as trash, survey information from 35 Inspection and Enforcement (I&E) inspectors indicated that 30-gallon trash containers are the most likely size of containers found at nuclear power plants [6]. Test data for fires of this size indicates that an energy output of 500 kW for 15 minutes represents what could reasonably be expected.
2. A pool fire (which most of the fire codes are designed to handle in a straightforward way and are benchmarked against) will be used as a second transient source. A spill of approximately 38 liters (10 gallons) of acetone spread out over an area of 4 square meters will be postulated. With the parameters used in the computer code, this fire results in an intensity of 4.67 MW for approximately three minutes. The total heat released in the fire is approximately $8.9E+08$ Joules. The volume of this spill was determined by examining several Appendix R submittals.

Typically, a liquid spill is postulated in these analyses near safety-related equipment and on the order of 38 liters is required to cause a problem based on utility analyses. Postulating a 38 liter spill that is of the same relative size as that used by the utilities provides a means to verify results by using a different calculational tool. Additionally, tests performed at Sandia to evaluate the 20-foot cable separation requirement of Appendix R required an equivalent amount of a liquid fuel to cause problems [7]. A 2-meter square pool was chosen to represent an "in between" size pool. Although the actual size would tend to be larger for an unconfined pool (a depth of about 1.75 mm based on the surface tension of acetone), the resulting fire would be very intense and of very short duration, probably insufficient to cause cable damage. On the other hand, a smaller pool would burn less intensely, but for a longer period of time. (A smaller pool is partially considered using the trash fire which burns at 500 kW for 15 minutes.) Because it is not known exactly how and where a liquid spill would occur, these two fires were chosen in an attempt to keep the analysis conservative. Electrically-initiated fires are not being explicitly evaluated, but it is assumed that the exposure fires postulated for this analysis bound any transient or electrically initiated fire that might occur at a plant.

In addition to the above fuel parameters, the worst case geometries are used in assessing fire growth and spread (e.g., most in situ fuel loading in an area, redundant cables pass the closest to one another, etc.). With these assumptions, the fire code COMPBRN was used to determine if the cables or components of interest would be damaged by the fire. This code has been used extensively by other fire PRAs and is designed to handle cable tray geometries. The code was used in a natural ventilation mode to evaluate the effects of the fire since its ability to handle a forced ventilation problem has not been adequately benchmarked to date.

- d. Arkansas Nuclear One-1 transient event trees are used to determine the systems/components, from a success oriented outlook, that need to be considered.

A success oriented approach was used since (1) if the "OK" states on the event tree cannot be negated, then a fire scenario cannot lead to a core melt (except with random failures considered), and (2) if it is shown that an "OK" state is vulnerable to a fire, the core melt probability has increased. By knowing what the vulnerable systems are, "modifications" can be recommended that significantly reduce the core melt probability. In this way, the "best" modification can be recommended rather than trying to fix everything. Transient event trees were used in this analysis since it was assumed that a fire occurs independently of any

initiating event and the operators will always insure that at least a transient will occur once they know that there is a fire (i.e., scram the plant). This approach is consistent with previous PRAs. Fires will not be assumed to start as a result of a transient. For LOCAs, the findings of previous PRAs have shown that, in general, the probabilities of LOCAs occurring as a result of or in combination with a fire are small. Therefore, those systems required solely to mitigate LOCAs will not be analyzed.

- e. Components affected by a fire are electrical/active in nature (e.g., cables, motors, switchgear, buses).

Electrical equipment is by and large susceptible to fires since a fire either changes the electrical properties of the components or physically causes damage (e.g., insulation damage, melting of contact points, shorts or opens). Although motors are not expected to be a critical failure component, they are assumed to fail in five minutes at 20 percent above their maximum manufacturers specified operating temperatures. Sensitive electrical components and those electronic systems typically found in electrical cabinets are assumed to fail at an internal cabinet ambient temperature of 150°F (based on values used in previous PRAs and a review of commercially available integrated circuit manufacturers catalogs). Buses will fail at a similar temperature since fuses and breakers might fail rendering the bus inoperative. Cables will be assumed to fail at their autoignition temperature or from time at temperature based on tests run at Sandia for cabling "cooked" at a constant temperature until failed [8]. The performance of the particular cable types used in the Arkansas Nuclear One-1 plant may vary from the cable tested at Sandia. However, it is assumed, without test proof, that the variations are insignificant. The time at temperature failure criteria are given in Table D.4 for qualified and unqualified cable.

Only temperature will be used to evaluate component vulnerability to a fire. Passive components, such as pipes and manual valves will be assumed to be unaffected by a fire.

Note: Other potential damage mechanisms, such as smoke, corrosive vapors, increased humidity due to the fire environment or due to suppression activity (except in a bounding sense for manual suppression) are not being used to evaluate component damage. These effects could have significant impacts on the core melt probability because they may affect components directly in the fire zone or in other locations (i.e., smoke could be transported via a ventilation systems or by other means to another zone and cause inadvertent suppression system actuation or direct equipment damage.

Thus a seemingly insignificant fire might cause significant damage.) There presently does not exist a data base that could be used to quantify these effects. In this analysis credit is given for an automatic suppression system actuating and not causing subsequent component damage. However, these automatic systems might in fact cause damage and the reader should keep in mind that this could only increase the core melt probabilities presented in the analysis section of this report. However the magnitude of this increase is unknown.

- f. Random failures of those systems not directly affected by the fire at a particular location will be considered if applicable.

For those cases where a particular safety train is not affected by the fire or where a remote shutdown capability exists, random failures of those systems left after the fire will be evaluated. These would be the systems or trains that for the short term would be expected to provide decay heat removal. This approach ensures that credit is given for these systems but that the potential for their incorrect operation is factored into the analysis (i.e., after a fire, a system could still randomly fail).

- g. Offsite power was assumed to be available during a fire unless the fire was capable of causing a loss of offsite power.

In most situations, it was assumed that given a fire and a reactor scram, offsite power would be available to the plant for recovery actions. This assumption only affects those sequences that include the power conversion system in one of the success paths on the event tree. Thus, all possible success paths that are indicated on the transient event trees were included in the analysis.

- h. Fire growth and spread and suppression of fires are treated as independent of each other.

Consistent with other fire PRAs, this assumption was made to make the analysis a workable problem. In a "real" fire, as suppression is applied, the fire would not instantaneously be extinguished, but would slowly decrease in flame size and energy output. However, there are currently no models available that can adequately address these two competing processes. Consequently, when suppression action takes place, the fire was assumed to be instantaneously put out.

- i. No explicit core melt timing was evaluated.

The TAP A-45 systems analysis includes thermal-hydraulic phenomenology in determining the timing for internal event sequences to determine potential recovery time windows. Typical time windows before core melt ensues are several hours. However, for the fire analysis, it was assumed that the repairs that would be required to be made after a fire had occurred could not be accomplished in these time windows because of the multitude of damaged cables and components that might be expected. Therefore, the core melt probabilities do not reflect heroic actions that might be taken by plant personnel to return some decay heat removal systems to operation prior to the onset of core melt. This assumption is a conservative approach and is consistent with previous fire PRA assumptions.

- j. Where exact locations of cables cannot be accurately determined, assumptions as to their locations are made.

Because of the time and effort that would be involved in tracing the locations of cables in say, a cable spreading room, simplifying assumptions are made as to the locations of redundant cables unless the documentation readily available shows the positions of these cables. For instance in the cable spreading room, a horizontal separation of 3 ft and a vertical separation of 5 ft for redundant cables will be used if no other information is available.

- k. Credit is given in the calculations for use of radiant energy shields, conduit, solid bottom cable trays, and completely enclosed cable trays.

Where radiant energy shields were used to prevent direct flame impingement, credit was given for the shields to the extent that only hot gas layer damage was considered where the cables would otherwise be in the plume directly above the flames. Also, rather than assuming that the cables autoignite at their piloted ignition temperature, they were assumed to ignite at their autoignition temperature. This latter assumption was also used if cables were in solid bottom cable trays. Where conduit or cable trays with solid tops and bottoms were used, the only credit allowed was that the cables would not contribute any fuel to the fire and that the cables would not be instantaneously damaged until they reached their autoignition temperature.

Note: The approach used in this analysis is designed to primarily look at the effects of a fire on a single unit. The results from the analysis will be presented for one unit, but if applicable, insights on how the fire could affect a two unit plant will also be included.

2. AREA ANALYSIS AND PROPOSED MODIFICATIONS

Note: This analysis assumes that all modifications proposed in the utility Appendix R submittals [9,10] have been or will be completed as stated.

2.1 Initial Screening to Determine Potentially Important Areas

2.1.1 Systems Analysis Data

This section was developed from the T3 transient event tree for Arkansas Nuclear One 1 and related fault trees and simplified piping diagrams. The following front line and support systems (success criteria are also listed) may be used in various combinations for a safe shutdown:

- a) Power Conversion System (PCS) 1 of 2 main feedwater pumps and 1 of 3 condensate booster pumps
- b) High Pressure Injection System (HPIS) 1 of 3
- c) Emergency Feedwater (EFW) 1 of 2
- d) Auxiliary Feedwater (AFW) 1 of 1 AFW pump and 1 of 3 condensate booster pumps
- e) Service Water (SWS) 1 of 3
- f) Emergency AC Power (available from offsite power or 1 of 2 Diesel Generators (DG) 1 of 2 distribution systems
- g) Normal offsite AC power
- h) DC Power (available from 1 of 3 battery chargers operating on AC power or 1 of 2 sets of station batteries) 1 of 2 distribution systems

The following combinations of the above systems may be used for safe shutdown:

- a) PCS using 1 of 2 turbine-driven main feedwater pumps and 1 of 3 motor-driven condensate booster pumps with steam being dumped to the condenser for long term heat removal. DC power and offsite AC power are required along with normal non-safety support systems for the PCS.
- b) EFW using 1 of 1 motor-driven pump or 1 of 1 turbine-driven pump. Feedwater is supplied from the condensate storage tank with the SWS acting as a backup supply of water. Emergency AC and DC power are required for either shutdown method, but use of the turbine-driven pump delays the requirement for AC power.

- c) AFW using 1 of 1 motor-driven auxiliary feedwater pump with 1 of 3 motor-driven condensate booster pumps. DC power and offsite AC power are required for this shutdown method.
- d) HPIS system using 1 of 3 motor-driven pumps for "feed and bleed" operation with 1 of 2 safety relief valves. Emergency AC and DC power are required for this shutdown method.

2.1.2 Individual Zone Analysis

In this section, each fire area is examined for redundant safe shutdown equipment. The utility's Appendix R submittal [9] and the plant visit were the primary means used to identify the safe shutdown equipment in a particular zone. Core melt frequencies are calculated in the next section for those areas which are shown to be vulnerable to fires. Both fire areas and fire zones are defined at Arkansas Nuclear One. For this analysis, fire areas are considered for safe shutdown equipment, with the smaller zones considered when redundant equipment is located within an area.

Each fire area is discussed individually below:

Area A--Auxiliary Building, Elevation 317'

This area contains only one division of the decay heat removal/low pressure injection system. All other systems would be available for a fire in this area, including the PCS.

Area B--Portions of Auxiliary and Turbine Buildings

This area consists of a large portion of the turbine and auxiliary buildings for both units. The redundant equipment of interest in this area is HPIS pump cables, SWS pump cables, and HPIS isolation valves. Also, PCS equipment is located in this area. Because the EFW system is independent of this zone, the only redundancy of concern in this area is the SWS pump cables. The SWS would be required to feed the EFW system once the condensate storage tanks for both units become depleted. Service water is also required to cool the DGs if offsite power is lost. Approximately 10 hours of EFW operation would be available from each unit's condensate storage tank. Use of the Unit 2 tank requires that a manual valve be repositioned, but sufficient time for this operation would be available. With the Unit 2 tank, an additional 10 hours of EFW operation would be available. Thus, a total of about 20 hours of EFW operation is possible before the SWS becomes necessary (assuming offsite power is available). Nevertheless, SWS cabling has been protected in this area by either a one hour rated fire barrier or more than 6.1 meters (20 feet) of separation with no intervening combustibles (Appendix R requirements). Automatic suppression is also installed in selected parts of this area.

Because of these protection measures along with the significantly delayed requirement for SWS (in which time many different actions could be taken), no further analysis of this area will be performed. It should be noted that a fire in this zone is assumed to not cause a loss of offsite power.

Area C--Auxiliary Building, Elevation 335'

The redundant equipment of concern in this area includes both of the EFW pumps, all the HPIS pumps, and all the SWS cables. Loss of the SWS could cause a loss of the PCS and AFW. Consequently, this area contains sufficient equipment to prevent safe shutdown if all the equipment were lost. Two of the three SWS pump power cables have been protected by a one hour rated fire barrier in this area. Consequently, PCS and AFW would remain available for a fire in this area. The HPIS pumps are located in individual concrete cubicles. A fire affecting one pump would not likely affect the remaining pumps. Further, cables to one of the EFW pumps is to be protected by a one hour rated fire barrier. Because of the extensive protection provided in this area, no further analysis of this area will be performed.

Area D--"Green" Diesel Generator Room

This area does not contain any redundant safe shutdown equipment.

Area E--South Switchgear Room

Following completion of proposed modifications, this area will not contain any redundant safe shutdown equipment. The specific modifications proposed are to allow the "swing" SWS pump and the "swing" HPIS pump to be powered from either safety bus independent of this area. The reason for these modifications is that the Technical Specifications allow one pump in each system to be out of service indefinitely. Without the modifications, a single fire could disable two pumps with the third pump indefinitely out of service. However, because of the modifications, no further analysis of this area will be performed.

Area F--South Battery Room

This area does not contain any redundant safe shutdown equipment.

Area G--Control Room and Cable Spreading Room

This area contains the control room (CR) and the cable spreading room. Because all front line and support systems have equipment located in both these areas, they are considered further in section 2.2.

Area H--"Red" Diesel Generator Room

This area does not contain any redundant safe shutdown equipment.

Area I--North Switchgear Room and North Electrical Penetration Room

The redundant equipment of interest in this area is cables to 2 of the 3 SWS pumps and 2 of the 3 HPIS pumps. Modifications to allow the "swing" pump for each system to be powered from either safety bus will ensure that only one pump would be lost in a fire in this area. Because of the modification and because of the discussion concerning Area B, no further analysis of this area will be performed.

Area J--Containment

This area is not evaluated for fire vulnerability in this study.

Intake Structure

This zone contains all three SWS pumps and associated valving. As discussed above, SW is required only after many hours when offsite power is available. Manual operation of all SWS valves is possible in this time period and two of the three SW pump power and control cables are protected by a one hour fire barrier in this zone. Also, the pumps are in individual cubicles separated by missile shields which act as radiant energy barriers and floor drains are available for preventing oil spill fires from propagating to adjacent pumps. Consequently, this area was not evaluated further.

Yard Area Manholes 1MH04 and 1MH06

This area contains redundant SW pump cables. Only two of the three cables are routed in this area. Because the Technical Specifications allow any one of the three SW pumps to be out of service for an extended period of time, these manholes have been filled with sand and one of the two pump power cables is planned to be moved out of these zones so that the sand may be removed. Thus, this area was not evaluated further.

Yard Area Manholes 1MH09 and 1MH10

This area contains redundant cabling associated with the Unit 1 DG fuel oil transfer pumps. This cabling would only be required if a loss of offsite power occurred together with the fire. A fire in this area does not have the potential to cause a loss of offsite power. Also, the Unit 2 fuel oil transfer pumps can be used for Unit 1. Thus, no further analysis of this zone was performed.

2.2 Quantification of Selected Zones

The only area remaining after the initial screening is Area G containing the CR (Zone 129-F) and the CSR (Zone 97-R).

2.2.1 Zone 129-F--Control Room

2.2.1.1 Evaluation of Potential Damage

The control room contains circuitry for all safe shutdown systems. An alternate shutdown is provided independent of the control room and an automatic Halon system is provided to protect the false floor and ceiling areas. The emergency system panels are located close together, but are separated by a fire barrier between panels C-16 and C-18. Further, the feedwater and condensate system panels are located separate from the emergency systems. Consistent with the original assumptions, this control room configuration will not be analyzed further.

2.2.2 Zone 97-R--Cable Spreading Room

2.2.2.1 Evaluation of Potential Damage

The cable spreading room (CSR) contains circuitry for all safe shutdown systems. An automatic water suppression system and a remote shutdown are provided for this area. The cables in this room are located in cable trays throughout the room at various elevations from the floor to the ceiling. Some cable trays also cross one another at various elevations. Because the exact locations of redundant cabling in the CSR are not known, assumptions concerning the location of certain cables are necessary. The location of equipment in the control room gives an indication of where the cables are located in the CSR. Based on the control room layout, a horizontal distance of 2.13 meters (7 feet) was estimated between the emergency systems cables and the PCS cables in the CSR. Consequently, these estimated distances were used in COMPBRN calculations. The COMPBRN calculations indicate that for the acetone fire, the redundant cables would ignite approximately two minutes into the fire.

2.2.2.2 Event Tree Development

The generic probability of a CSR fire is $7.0\text{E-}04/\text{Rx yr}$ (median) or $6.7\text{E-}03/\text{Rx yr}$ (mean). Using the mean value for fires in the CSR yields a room probability for fire of $6.7\text{E-}03/\text{Rx yr}$.

A water suppression system is installed in the CSR; the generic unreliability of this system is 0.04/demand. In the event of automatic suppression failure, manual suppression may still succeed. The generic probability of nonsuppression in five minutes (the approximate time to damage of redundant trains) is 0.4/demand.

The probability of failure to correctly align and start the primary remote shutdown system (EPW) is taken as 0.1/demand. This value is based on Seabrook data for the conditional frequency that the operators will not do the right thing at auxiliary shutdown panels. Once the system is running, the probability of failure is taken as 0.01/demand. In cases where the operator has failed to start the primary remote shutdown system or the primary remote shutdown system has failed, a backup shutdown method is available using the HPIS system in a "feed-and-bleed" mode of operation. The probability of successfully aligning and starting the secondary shutdown method, given failure of the primary shutdown system, is taken to be 0.5. Once the backup system is started, the probability of random failures in the system is taken as 0.01.

An event tree for the CSR is shown in Fig. 1 with a resulting core melt probability from fires in the CSR being $5.8\text{E-}06/\text{Rx yr}$.

2.3 Proposed Modifications and Modified Event Trees

2.3.1 Cable Spreading Room

Potential modification #1: Protect one division of the ESF cable trays and cables in the CSR with a one hour rated fire barrier to prevent damage to the cables from exposure fires. The fire barrier would also prevent the protected cables from contributing fuel to postulated fires.

The core melt probability reduction resulting from this modification would incorporate random failures of several automatic systems with all normal power and instrumentation intact for the protected division. Protection of one division would lower the core melt probability by several orders of magnitude. In reality, if only one fully automatic system and all its associated power, control, and instrumentation were protected, the core melt probability could be reduced well

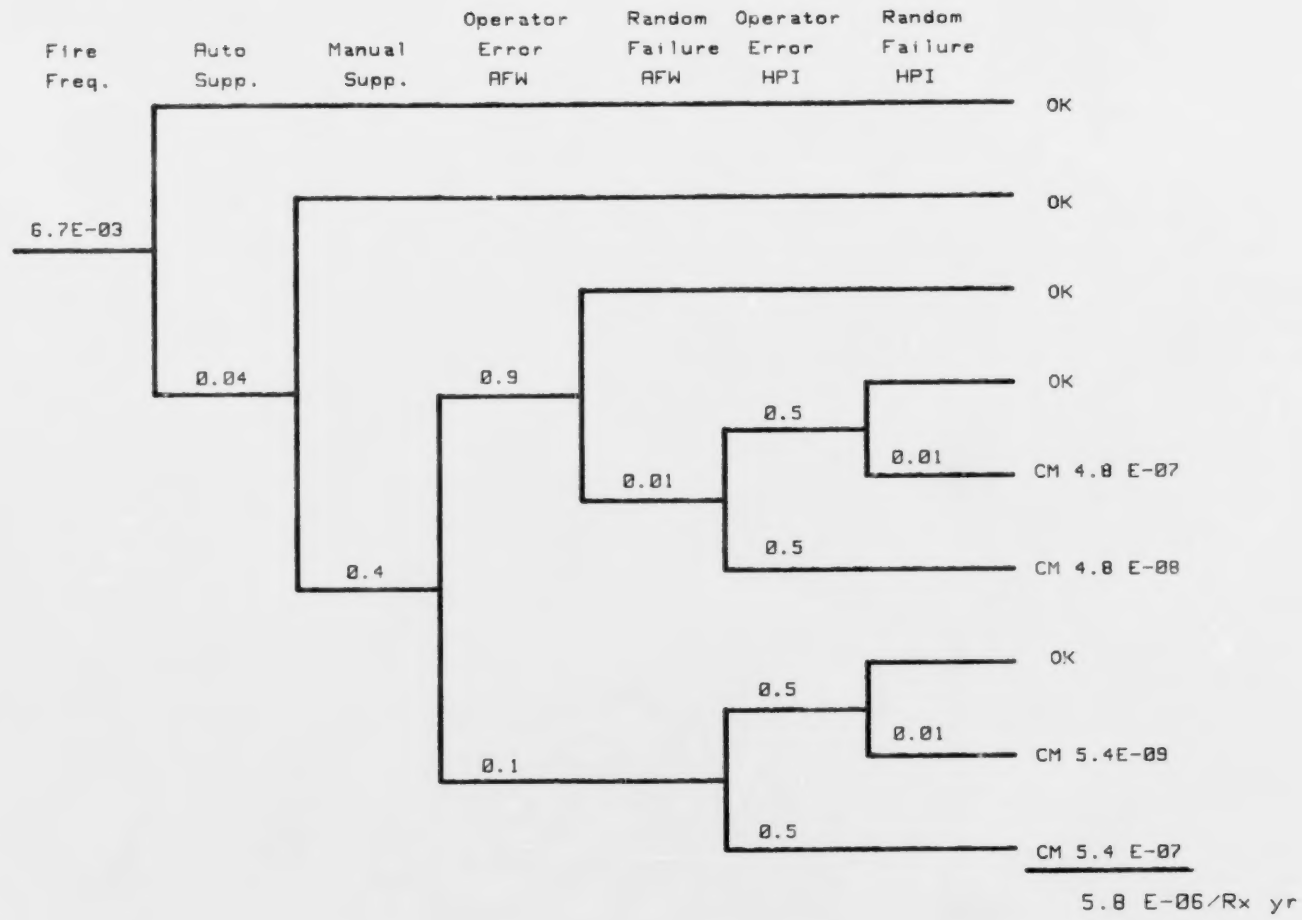


Figure 1: Event Tree for ANO Cable Expansion Room

below the cutoff value of 1.0 E-06/Rx yr. However, identifying and locating all the cables of a single system together with all associated cables which might prevent correct operation of the single system could be very difficult. Protecting all the cables of one division may also be an extremely difficult task considering the number of cables in the CSR and the limited access in the CSR.

Potential Modification #2: Install a redundant, independent, automatic suppression system in the CSR to improve the reliability of the automatic suppression system.

The most reliable system which could be made completely independent of the water suppression system would be a CO_2 system. Fig. 2 is a modified event tree for this modification with a resulting core melt probability of 5.8E-07/Rx yr. The probability of failure of the automatic systems has been modified to reflect the improvement made by installing the backup system. The probability of failure of the CO_2 system has been taken as 0.1 from Table 2.

Proposed Modification #3: Install redundant detection, piping, and valving for the current water suppression system to improve its reliability.

The redundant equipment would require independent power and would require valving and piping which would minimize the potential for common mode failures resulting from improperly positioning any single valve (especially manual valves). The probability of failure of the modified automatic suppression system is taken as the product of the independent probabilities plus the probability of a common mode failure. The common mode failure probability is estimated as one order of magnitude less than the independent system reliability, consistent with probabilistic risk assessment practice when specific data is not available. Thus, the probability of failure of automatic suppression is $(0.04 \times 0.04) + 0.004 = 0.0056$. Using this value in an event tree identical to Fig. 1 gives a core melt probability of 8.1E-07/Rx yr.

Beyond the above potential modifications, cables could be rerouted out of this room to make one automatic system independent of the CSR. No quantification of potential benefits of this modifications was performed.

3. CONCLUSIONS AND INSIGHTS

Only one location at Arkansas Nuclear One 1, the CSR, was identified as being vulnerable to fires with a core melt probability greater than 1.0E-06/Rx yr. The core melt probability for the CSR was calculated as 5.8E-06/Rx yr. Protecting one division of cable trays with a one hour

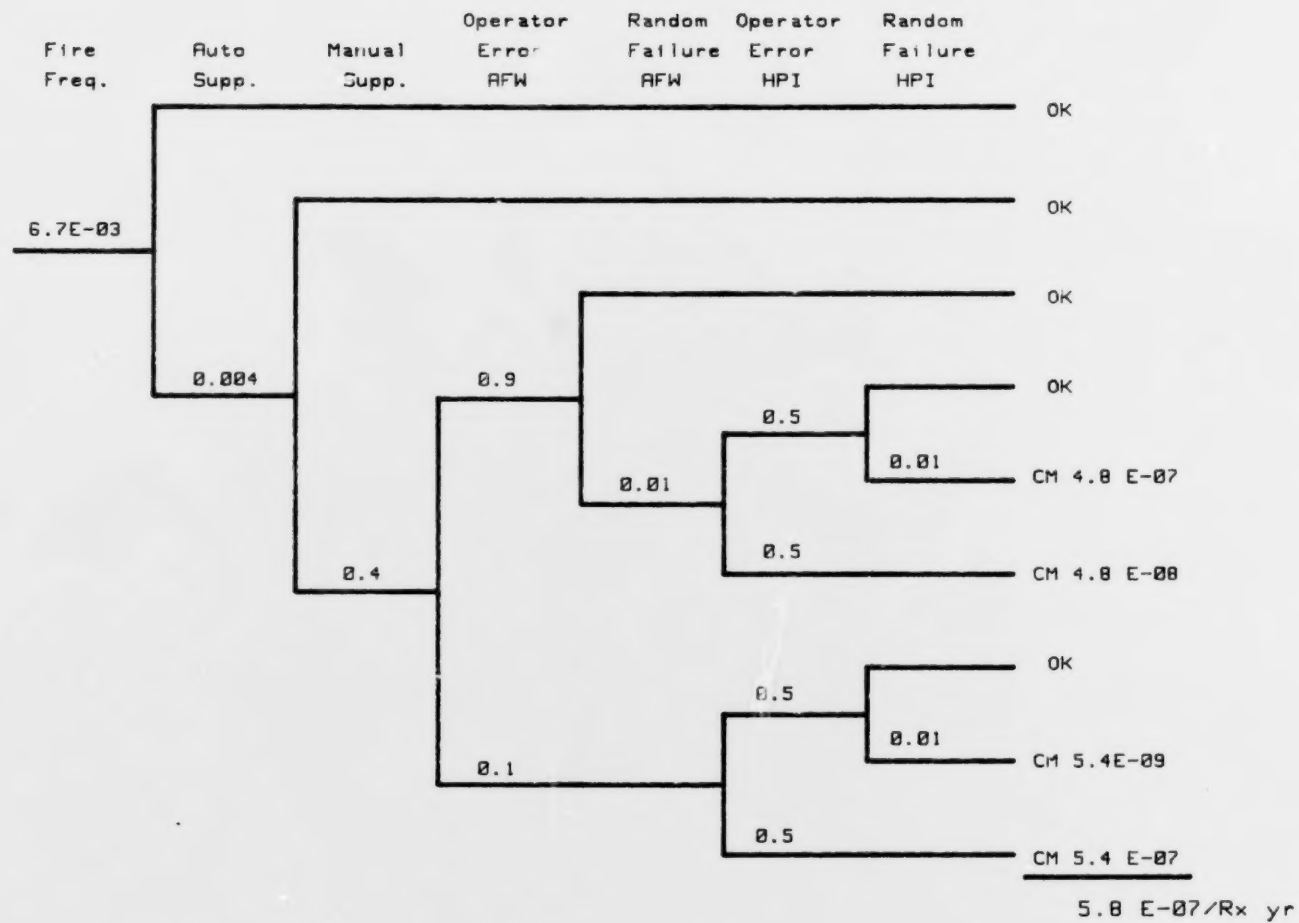


Figure 2: Modified Event Tree for ANO Cable Expansion Room

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rated fire barrier would result in a core melt probability reduction to well below the cutoff core melt probability of $1.0\text{E}-06/\text{Rx yr}$. Installing an automatic CO_2 system as a backup to the water suppression system would lower the core melt probability estimate to $5.8\text{E}-07/\text{Rx yr}$; incorporating improvements to the current water suppression system would lower the core melt probability estimate to $8.1\text{E}-07/\text{Rx yr}$. Because of the anticipated difficulty with wrapping cables in the CSR, the most viable modification appears to be the installation of a CO_2 suppression system or improving the reliability of the current water suppression system.

Insights: Specific plant and uncertainty insights are presented below:

Plant Insights:

1. The CRs and CSRs are not shared between the units, and no other areas were identified where a fire has a high likelihood of affecting both units simultaneously.
2. Several fire zones not quantified in this analysis have more than one train of front-line safety systems or essential support systems. However, redundant systems or effective protection of the redundant trains was provided.
3. The cable spreading room was quite congested with cables and cable trays running in many different directions. The assumptions concerning the locations of redundant cables in the cable spreading room are not expected to significantly affect the COMPBRN calculation results for the given initiating fires.

Uncertainty Insights

Some of the potential areas of uncertainty are identified as follows:

- a. Effects due to suppression activities (both automatic and manual) on safety equipment. This includes suppression effects in the zone where the fire is as well as the effects in other areas.
- b. Effects due to smoke, corrosive gases and humidity changes caused by a fire on safety related equipment.
- c. The method of partitioning used to determine generic room or building fire frequencies from the fire occurrence data base. Similarly, the applicability of using generic fire frequency data representing all plants for an individual plant.

- d. Manual suppression probabilities relating the time available prior to component damage to the probability of not being able to suppress the fire in the given time frame.
- e. Capability of deterministic codes to accurately predict the actual fire environment seen by safety equipment when a fire occurs.
- f. Electrical independence between control room circuits and remote shutdown type panels (either one central panel or several small panels located in different zones) that allow for control of safety equipment if the control room is evacuated.

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5. TABLES

Table D.1: Fire Frequencies for Various Nuclear Power Plant Locations [3]

Location	Mean Fire Frequency
Control Room (per year)	4.9E-03
Cable Spreading Room (per year)	6.7E-03
Auxiliary Building (per year)	4.8E-02
Turbine Building (per year)	1.6E-02

Table D.2: Probability of Automatic Suppression Systems Failing to Function [2]

Type of Suppression System	Probability of Failure to Function(per demand)
Halon	0.2
Carbon Dioxide	0.1
Water	0.04

Table D.3: Manual Nonsuppression Probabilities as a Function of Available Time [3]

Time Available for Suppression	Probability of Nonsuppression
5 minutes	0.4
15 minutes	0.3
30 minutes	0.2
60 minutes	0.1

Table D.4: Cable Failure Time vs. Temperature for Qualified and Unqualified Cables [6]

Time to Failure (minutes)

Temperature (°C/K)	Unqualified Cable	IEEE-383 Qualified Cable
250/523	7	Not tested
350/623	4	13
450/723	2	4
500/773	Not tested	3

Appendix E

INTERNAL FLOOD RISK ANALYSIS FOR
ARKANSAS NUCLEAR ONE-1 NUCLEAR POWER PLANT

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APPENDIX E INTERNAL FLOOD RISK ANALYSIS FOR ARKANSAS
NUCLEAR ONE-1 NUCLEAR POWER PLANT

1. ANALYSIS METHODS AND ASSUMPTIONS

1.1 Introduction

Operating experience at nuclear power plants has shown that the potential for damage to safety-related equipment as a result of internal flooding is of concern. Examples of how plants have experienced internal flooding include operators overfilling water tanks, in-leakage of ground water, hose and weld ruptures, pump seal leaks, improper maintenance procedures, and various other circumstances. Several of these events resulted in damage to high pressure coolant injection systems, reactor building spray pumps, service water pumps, emergency feedwater pumps, diesel generator control cabinets and other redundant and diverse safety-related components. Although to date, none of these floodings have led to a serious accident, there is still a risk that needs to be evaluated to determine the probability that an internal flood, despite a conscious effort to design the plant to withstand this threat, could lead to a core melt scenario. The TAP A-45 program specifically addresses these concerns for the decay heat removal systems and the sections that follow will present the methodology and assumptions employed to carry out the internal flood analysis.

1.2 Scope

- a. This analysis will evaluate for the Arkansas Nuclear One-1 nuclear power plant the risk from internal flooding that could affect decay heat removal systems. The results of this analysis, the quantified risk and any modifications that may be proposed, will be compared to risk from other initiating events with the modifications evaluated for risk reduction and cost. This "value impact" analysis will then ultimately determine the optimum set of modifications to get the desired reduction in risk.
- b. The analysis is not intended to be a comprehensive internal flood PRA. An analysis of this sort would be beyond the scope of TAP A-45. The intent of this analysis is to quantify significant internal flood threats to a plant. In doing so, existing procedures used in published PRAs will be employed, but simplifying assumptions will be made to allow the analysis to proceed without requiring the level of detail required for these other PRAs. In that respect, this analysis embodies the basic approaches used by previous internal flood PRAs but is by necessity a scaled down version of these PRAs.

- c. Wherever possible, data from previous PRAs will be used. Since this analysis embodies what has been done in previous PRAs, use of this data will provide consistency in how the data is used and allow some comparison between the analyses. The data is based primarily on that found in the Seabrook internal flood PRA [1].
- d. Because this analysis is a scaled down version of an internal flood PRA, the quantification of the risk due to an internal flood will be presented as a point estimate. Uncertainty values will not be formally determined. However, insights into the potential major areas of uncertainty will be identified.
- e. The systems to be examined and quantified are based on the event and fault trees developed by the systems analysts, and is thus constrained by the level of detail of the systems analysis effort. In a similar fashion, readily available plant documentation determines the level of physical detail used in this analysis. Documentation obtained for this effort is treated as accurate and up to date unless otherwise identified.
- f. A screening cutoff of $1.0E-06$ for an internal flood-induced core melt probability is used in the analysis to determine if further quantification efforts in an area are warranted.

This is based on insights from previous PRAs where the internal events analysis lead to core melt probabilities on the order of $1E-04$. Any internal flood produced core melt probabilities less than $1E-06$ would therefore be insignificant compared to the internal events.

1.3 Analysis Methodology

The approach used in determining the risk due to internal flooding at the Arkansas Nuclear One-1 nuclear power plant used the following four basic steps:

- a. Initial screening for potentially important internal flood levels
- b. Plant visit for verification
- c. Quantification
- d. Proposed modifications to reduce the risk due to internal flood in affected areas

1.3.1 Screening for Potentially Important Internal Flood Areas

The purpose of this step was to determine which levels in the plant have "enough" of the front line or support systems in their boundaries such that internal flood damage could invalidate all the success branches on the transient event trees.

The sequence of steps taken during this first screening were as follows:

- a. Internal event analysis data was obtained as the starting point. The specific data included event trees, fault trees, and simplified piping diagrams used in constructing the fault trees.
- b. The transient event trees were examined to determine those front line systems or combinations of systems that would lead to an 'OK' state on the event tree. Also obtained from the event trees were the success criteria of the front line systems.
- c. The next step was to identify the support systems required for the front line systems identified above. Support systems included cooling systems and AC/DC power requirements. The success criteria for the support systems were obtained from the systems analysis. For all systems except the power conversion system, the fault trees for the front line systems were used to determine the support systems. For the power conversion system, plant schematics and piping diagrams were used to determine required support systems.
- d. Once the systems of interest were identified, all available documentation (principally FSARs and plant supplied information and drawings) were analyzed to determine those areas where "enough" components could be disabled by an internal flood. Based on this screening, a small set of areas were left for further analysis.

To this point in the analysis, everything was based on a review of available documentation.

1.3.2 Plant Visit

A visit was made to the plant with the following objectives:

- a. Based on the initial screening, the general location of components of the systems of interest are known. The next step was to visit the plant and see the physical

arrangements in each of these areas. The plant visit also provides a means to determine locations of major flood sources and some estimate of the x, y, z coordinates of components. These locations are necessary for the quantification step.

- b. The second purpose of the plant visit was to confirm with plant personnel that the documentation that was being used was in fact the best available information and to obtain clarification about any questions that may have arisen in a review of the documentation.

1.3.3 Quantification

For those areas remaining after the initial screening, the next step was to determine if in fact a flood source could damage the components of interest in the area, and if so, how much time would be available for operator action before all "OK" states were negated on the event trees. This calculation was based on Standard Review Plan 3.6.1 and 3.6.2 for determining break sizes for moderate energy lines. With the flow rate, room volume, and capability of water removal known, the time to damage of components was calculated based on the water depth in the room and the assumptions governing component damage thresholds. Spray was considered when a flood initiates in a room and the critical components are in a direct line of sight with the flood source. It was assumed that these components failed due to the spray; other critical components that were either in the same room or in different rooms could fail due to submergence caused by the flood source. Based on the results of the analysis, further quantification was either continued or the room was determined not to be susceptible to internal flooding (subject to the assumptions and limitations of the analysis).

An event tree was utilized to estimate the risk of core melt from internal flood/spray in those areas identified in the initial screening and not eliminated by subsequent analysis. The basic elements of the event tree are as follows:

- a. Internal flood occurrence frequency for the area of interest.
- b. Isolation potential of the flood source.
- c. Operator error and random failures of unaffected systems.

Each of these elements is discussed below:

- a. Internal flood occurrence frequency for the area of interest.

The internal flood frequency was based on operating experience data and is related to the size of the flood source. The flood frequencies used in this analysis are given in Table E.1. [2]

Table E.1: Internal Flood Frequency as a Function of Location and Severity

Location	Severity Level	Mean Flooding Frequency (per reactor year)
Turbine Building	Moderate	1.05E-02
Turbine Building	Large	8.1E-03
Turbine Building	Very Large	7.7E-03
Auxiliary Building	Small	4.4E-03
Auxiliary Building	Moderate	2.2E-02
Auxiliary Building	Large	1.2E-02

b. Isolation potential of the flood source

Source Isolatable from the Control Room - This step assigns a probability that the operator will correctly isolate the flood from the control room. If the flood is not isolatable from the control room, this probability is set to zero. The probability is determined by using the time to submergence of the critical components and determining from human factors considerations the probability that the operator takes the proper isolation actions.

Note: Responses on control room instrumentation that alert the operators to a potential flood problem will not be considered since a) it is not clear how these instruments would respond and display their information and b) the complex sequence of events that might take place would require detailed system and human reliability modeling efforts, which is beyond the scope of TAP A-45.

Isolatable from the Plant - This step is similar to the above step, except that the human factors probabilities will be modified to reflect a more stressful situation since the operators would be assumed to be manually repositioning valves in the plant.

c. Operator error and random failures

An evaluation is made of operator error and random failures for any systems that are left unaffected by the internal flood. Specifically, any systems that could be used to provide a decay heat removal function are evaluated.

If manual operator action is required at a remote location, the probability of operator error is taken as 0.1. If a system can be operated from the control room, the probability of operator error is taken as 0.05. If the system is fully automatic, the random failure of a single train is taken as $1.0E-02$.

1.3.4 Proposed Modifications

Based on the quantification steps, engineering modifications were proposed for those areas which have an internal flood-induced core melt probability contribution of greater than $1.0E-06/\text{Ry}$. Proposed modifications are of the following nature:

- a. Installation of automatic flood detection systems.
- b. Construction of flood barriers to isolate portions of an area.
- c. Physically moving redundant system hardware to other locations to prevent its loss during an internal flood.

For any proposed modification, a core melt probability was recalculated to indicate how much the core melt probability could be reduced. The quantification was performed by modifying the internal flood event tree to reflect the modification

The proposed modifications were supplied to an architectural firm for cost evaluation during a second plant visit. A "value impact" analysis which considered the core melt contributions from internal floods and other external and internal events was used to determine the "best modification" in an overall sense.

1.4 Assumptions

Most of these assumptions are based on insights derived from previous PRA's or by consideration of the physical constraints of a flood in a building. The major assumptions are as follows:

- a. Internal flood frequencies, component damage thresholds, and physical room considerations will be based on previous industry wide data bases.

Existing data is used to the extent possible to evaluate the threat of an internal flood to a plant. Internal flood frequencies have been determined based on operating experience. These frequencies have been determined by the amount and flow rate from a flood source. For instance the Seabrook PRA has defined four levels of flooding:

1. Small - flooding on the order of hundreds of gallons, e.g. valve pit flooding, instrument flooding, or flooding within a component.
2. Moderate - flooding on the order of several thousand gallons, e.g.; a few feet of water on the floor of a typical pump room.
3. Large - flooding on the order of tens of thousands of gallons; e.g; a few feet of water in large rooms, very deep water (more than ten feet) in a typical pump room.
4. Very large - flooding on the order of hundreds of thousands of gallons, e.g. floods involving circulating or service water piping.

Other attempts at quantifying internal flood frequencies have considered the flow rate from the flood source.

The drainage available is based on an equivalent diameter approach (i.e. a square, 1 ft by 1 ft opening would be considered capable of handling the flow from a pipe with a diameter of approximately 1 foot).

Previous internal flood PRA's have used a factor of 20 percent to derate the floodable volume of a room. This accounts for equipment taking up some of the area. This analysis will use the same value.

- b. Arkansas Nuclear One-1 transient event trees are used to determine the systems/components, from a success oriented outlook, that need to be considered.

A success oriented approach was used since 1) if the "OK" states on the event tree cannot be negated, then an internal flood cannot lead to a core melt (except with random failures considered), and 2) if it is shown that an "OK" state is vulnerable to an internal flood, the core

melt probability has increased. By knowing what systems are vulnerable, "modifications" can be recommended that significantly reduce the core melt probability. In this way, the "best" modification can be recommended rather than trying to modify everything. Transient event trees are used in this analysis since it is assumed that an internal flood occurs independently of any initiating event and the operators will always insure that at least a transient will occur once they know that there is an internal flood (i.e., scram the reactor). This approach is consistent with previous PRAs. Internal floods will not be assumed to start as a result of a transient. LOCA event trees will not be examined since pipe ruptures have already been considered in these trees. In addition, the findings of previous PRAs have shown that, in general, the probabilities of LOCAs occurring as a result of or in combination with internal flooding are small. Therefore, those systems required solely to mitigate LOCAs will not be analyzed. As an extension of this assumption, LOCA analyses will be assumed to bound all internal flooding which could occur inside primary containments and this analysis will exclude primary containment flood sources.

- c. Front line mitigation systems (such as high and low pressure injection) will not be considered as flood sources.

The rationale behind this assumption is that front line systems (a) have been designed to higher standards (seismically qualified, and extensive material and workmanship quality assurance standards), (b) have controlled chemistry, (c) do not see large temperature gradients in general, (d) have been involved in few catastrophic failures, and (e) are not normally operating. Consequently, the chance of these systems being a flood source is minimal. Further, most front line systems only have a limited volume of water from which to draw suction. Typically this source is a tank. Because this analysis considers potential flooding in each zone from tanks emptying their contents into a single room, this is in essence no different than considering that a front line system pipe ruptures and runs until it completely empties its source (a tank). Therefore the tank analysis effectively bounds potential internal flooding from front line systems.

High energy fluid systems (i.e., greater than 200°F and 275 psig) will not be considered because of the following probabilistic and physical considerations:

1. The break of a high energy line, while it might cause damage to those components in its immediate area, would not be able to damage all safety systems of interest due to spray since plants in general are designed such that redundant safety systems are not in the same physical locations. Therefore, the probability of leading to core melt would be the probability of the break occurring times the random failure of the systems left. Typical numbers for these values respectively are $1.0E-2$ and $1.0E-5$. Thus the core melt probability would be on the order of $1.0E-7$ which is below the cutoff being used in this analysis.
2. Plants are required to verify that the design of components is sufficient to withstand the most severe environment that they could be expected to see. Therefore the environmental conditions that could be expected from a high energy line break are assumed to not affect components.

(Note that the above is assumed to be true in this analysis; however, as pointed out elsewhere in this analysis, this may not be true.)

d. Flood barriers maintain their structural integrity.

This analysis will not determine the hydrostatic loading placed on walls due to a flood nor assess when and how these walls will fail. Because the walls at nuclear power plants are typically reinforced concrete and one to three feet thick, it is felt that before a wall would collapse from flood pressure exerted on it, the component(s) of interest would be submerged. Structural integrity also includes penetration seals and watertight doors.

In the cases of doors, the door frame is assumed to remain intact. Doors are treated explicitly in this analysis in that if the flood source seats a non-watertight door against its frame, the door is assumed to stop all but 10 percent of the flow out of the flood source in the room. Thus, a water level may be built up in the room that can potentially submerge a component. Conversely, if the flood source pushes against a door such that the latch is the only thing that is keeping the door shut, then it is assumed the door immediately opens and passes flood waters.

Note: Watertight doors and other doors are assumed to have a low probability of being open since these doors tend to serve as fire barriers and as such are designed to

be self shutting and/or are alarmed indicating their status.

- e. Flooding via drains backing up into other areas will not be considered.

A detailed analysis of plant drain systems to ascertain and evaluate the chance of drains backing up and causing problems will not be performed. (This would entail knowing flow paths, check valves installed, reliability of these valves, all possible sources that could feed combinations of these drain paths, etc.) However, one aspect of drains will be addressed. In all probability, when a flood occurs where there is a drain, some debris will eventually block all or a portion of the drain.

When and how much blockage would actually occur is unknown, but for this analysis it will be assumed that at the inception of a flood, drains are blocked by a factor of 20 percent of their flow rate. (i.e. the drain can handle only 80 percent of its design capacity) It is also assumed that the location of the ultimate drain source (e.g.; a tank or reservoir) is capable of handling the influx of water and does not become a secondary flood source.

- f. Only single flood sources will be analyzed.

Although it is recognized that a seismically-induced internal flood could cause multiple pipe breaks to occur in the plant, evaluation of this situation is beyond the scope of TAP A-45. This analysis is limited to single flood sources since they can be examined in a timely and efficient manner and are reproducible and tractable.

- g. Components will fail from submergence in water to a critical point or from direct spray from the flood source.

Electrical equipment (primarily motors/pumps, switchgear, and buses) is by and large susceptible to a flood since it causes electrical shorting of the component. Humidity will not be considered as a failure mechanism due to the lack of failure data available and because equipment is assumed to be qualified to the worst environment it will see. (Note: The purpose of TAP A-45 is not to experimentally develop this type of data.) Most purely mechanical components (i.e. those not using electricity as a motive force) are not directly vulnerable to water unless a) the water can cause mechanical damage, such as entering the casing of a turbine and causing blade damage or b) the control systems for these components are

electrical and their failure causes the component to malfunction. Mechanical component damage is considered when steam-driven pumps are important front line or support system components.

It is assumed that switchgear and buses will fail if they are standing in six (6) inches of water or more (typically contacts in large breakers are found at this level).

In a similar fashion, an electric motor will be assumed to fail when the water level reaches the bottom of the motor casing, unless it can be determined that the motor is designed to be watertight. Cables will be assumed to maintain their integrity if submerged. Consideration of motor operated valves (MOVs) is beyond the scope of TAP A-45 because of the large number of MOVs and the possibility of complex interactions and timing in valve alignments. These assumptions are consistent with those made in previous internal flood PRAs.

- h. Random failures of those systems not directly affected by the internal flood at a particular location will be considered if applicable.

For those cases where particular systems are not affected by the flood, random failures of those systems will be evaluated. These would be the systems or trains that for the short term would be expected to provide decay heat removal. This approach ensures that credit is given for these systems but that the potential for their incorrect operation is factored into the analysis (i.e., after an internal flood, a system could still fail randomly).

- i. Offsite power is assumed to be available during an internal flood unless the flood is capable of causing a loss of offsite power.

In most situations, it was assumed that given an internal flood and a reactor scram, offsite power would be available to the plant for recovery actions. This assumption only affects those sequences that include the power conversion system in one of the success paths on the event tree. Thus, all possible success paths that are indicated on the transient event trees are included in the analysis.

- j. No explicit core melt timing is evaluated.

The TAP A-45 systems analysis includes thermal-hydraulic phenomenology in determining the timing for internal event sequences to determine potential recovery time windows. Typical time windows before core melt

ensues are several hours. However, for the internal flood analysis, it is assumed that the repairs that would be required to be made after a flood had occurred would not be able to be accomplished in these time windows because of the multitude of damage that might be expected. Therefore, the frequencies that are determined from this analysis do not reflect heroic actions that might be taken by plant personnel to return some decay heat removal systems to operation prior to the onset of core melt. This assumption is a conservative approach.

Note: The approach used in this analysis is designed to look primarily at the effects of an internal flood on a single unit. The results from the analysis will be presented for one plant, but if applicable, insights on how the internal flood could affect a two unit plant will also be included.

2. AREA ANALYSIS AND PROPOSED MODIFICATIONS

2.1 Initial Screening to Determine Potentially Important Areas

2.1.1 Systems Analysis Data

This section was developed from the T3 transient event tree for Arkansas Nuclear One 1 and related fault trees and simplified piping diagrams. The following front line and support systems (success criteria are also listed) may be used in various combinations for a safe shutdown:

- a) Power Conversion System (PCS) 1 of 2 main feedwater pumps and 1 of 3 condensate booster pumps
- b) High Pressure Injection System (HPIS) 1 of 3
- c) Emergency Feedwater (EFW) 1 of 2
- d) Auxiliary Feedwater (AFW) 1 of 1 AFW pump and 1 of three condensate booster pumps
- e) Service Water (SWS) 1 of 3
- f) Emergency AC Power (available from offsite power or 1 of 2 Diesel Generators (DG)) 1 of 2 distribution systems
- g) Normal offsite AC power
- h) DC Power (available from 1 of 3 battery chargers operating on AC power or 1 of 2 sets of station batteries) 1 of 2 distribution systems

The following combinations of the above systems may be used for safe shutdown:

- a) PCS using 1 of 2 turbine-driven main feedwater pumps and 1 of 3 motor-driven condensate booster pumps with steam being dumped to the condenser for long term heat removal. DC power and offsite AC power are required along with normal non-safety support systems for the PCS.
- b) EFW using 1 of 1 motor-driven pump or 1 of 1 turbine-driven pump. Feedwater is supplied from the condensate storage tank with the SWS acting as a backup supply of water. Emergency AC and DC power are required for either shutdown method, but use of the turbine-driven pump delays the requirement for AC power.
- c) AFW using 1 of 1 motor-driven auxiliary feedwater pump with 1 of 3 motor-driven condensate booster pumps. DC power and offsite AC power are required for this shutdown method.
- d) HPIS system using 1 of 3 motor-driven pumps for "feed and bleed" operation with 1 of 2 safety relief valves. Emergency AC and DC power are required for this shutdown method.

2.1.2 Individual Zone Analysis

In this section, fire areas are used as a basis for determining if a given area has a sufficient amount of safe shutdown equipment that an internal flood could prevent safe shutdown. The utility's Appendix R submittals [3,4] and the plant visit were the primary means used to identify the safe shutdown equipment located in a particular area.

Area G--Control Room and Cable Spreading Room

This area contains the control room (CR) and the cable spreading room. Because all front line and support systems have equipment located in both these areas, they are considered further in subsequent sections of this analysis.

Area H--"Red" Diesel Generator Room

This area does not contain any redundant safe shutdown equipment.

Area I--North Switchgear Room and North Electrical Penetration Room

The only redundant equipment of interest in this area is cables. Since flooding is assumed to not damage cables, no further analysis of this area will be performed.

Area J--Containment

This area is the containment building and is not evaluated for internal flooding in this study.

Intake Structure

This zone contains all three SWS pumps and associated valving. The SWS is required only after many hours when offsite power is available. The intake structure contains numerous large diameter pipes, primarily associated with service water. A break in one of these pipes could cause some flooding of the 354' elevation. However, this elevation contains only piping, cabling, and valves associated with the SWS. The valves have stems extending up to the upper elevation of the intake structure (366') for flood protection and the SW pump motors are also located at the upper elevation. Consequently, this area is not considered further.

Yard Area Manholes 1MH04 and 1MH06

This area contains only cables. Consequently, it is not analyzed further.

Yard Area Manholes 1MH09 and 1MH10

This area contains only cables. Consequently, it is not analyzed further.

2.1.3 Evaluation of Combinations of Areas

In this section, combinations of areas on each plant level whose flooding could lead to core damage accidents are considered.

2.1.3.1 Elevation 317'0"

This level contains both decay heat removal vaults. Each is isolated from the other and from the general access area by flood-tight doors. Further, safe shutdown can be achieved without the decay heat removal system. Consequently, this level is not evaluated further.

2.1.3.2 Elevation 335'0"

Internal flooding of all of this level has the possibility of damaging the PCS, HPIS, AFW, and EFW. Consequently, this level is considered further in section 2.2.

2.1.3.3 Elevation 354'0"

This elevation does not contain any redundant equipment required for safe shutdown and is therefore not considered further.

2.1.3.4 Elevations 368'0" - 374'6"

This elevation contains normal and emergency 4160 VAC switchgear. Loss of all this switchgear could potentially prevent safe shutdown. However, open stairwells outside each of the emergency 4160 VAC switchgear rooms would provide drainage to lower levels as would an open hatch and several stairwells in the vicinity of the normal 4160 VAC switchgear. Flood sources outside the emergency switchgear rooms would therefore tend to run toward lower elevations. The only flood source potentials inside the emergency switchgear rooms are service water piping to the ESF room coolers and the normal room coolers located in the rooms. Except for a very small segment of the piping where it enters the coolers, the active piping (approximately 2" in diameter) is protected by an outer guard pipe (approximately 6" in diameter) along the length of the pipe. Because of the drainage available in the vicinity of the switchgear and the separation between the normal and emergency switchgear, this level is not considered further.

2.1.3.5 Elevation 386'0"

The only redundant equipment of concern on this level is located within the control room. A stairwell outside of one entrance to the control room and a hatch outside the other entrance would provide drainage for potential internal flooding. Further, the control room does not contain any significant internal flood sources and it is continuously manned so that flood detection and mitigation would be prompt. For these reasons, this level is not considered further.

2.1.3.6 Intake Structure

The intake structure is a separate building which was considered in section 2.1.2. No other areas are considered to be flooded simultaneously by a single internal flood source. Consequently, the intake structure is not considered further.

2.1.4 Summary of Initial Screening

Based on the initial screening, no individual areas were identified that might be vulnerable to internal flooding.

Only one level, elevation 335'0", was identified as having a sufficient amount of equipment that safe shutdown could be prevented. All other levels either do not have a sufficient amount of equipment or drainage to lower levels through hatches or stairwells would prevent the redundant equipment from being damaged. Elevation 335'0" is located below grade and is the elevation where water would accumulate if internal flooding were to occur at higher elevations.

2.2 Quantification of Selected Zones

Based on the initial screening, only one elevation, 335'0", needs to be considered further. No individual areas were identified that need to be considered further.

2.2.1 Elevation 335'0"

2.2.1.1 Evaluation of Potential Damage

Internal flooding of all of this level has the possibility of damaging the PCS, HPIS, AFW, and EFW. The PCS and AFW could be disabled because all the condensate booster pumps are located at this elevation in the turbine building. The EFW and HPIS could be disabled because the respective pumps are all located at this elevation in the auxiliary building. An internal flood would thus have to affect both the auxiliary building and the turbine building. These two buildings are separated by fire walls at this level, but sources from above this level could possibly flood both areas simultaneously. The level that would have to be flooded is approximately 4200 m² (45000 ft²). An estimated water level of 0.46 meters (1.5 feet) would be required to damage all the pumps. Reducing the floodable volume of this area by 20% leaves a required water volume of 1,530,000 liters (405,000 gallons) to cause damage to all the redundant pumps. Also, the internal flood source must be above this elevation and drain to both the auxiliary building and the turbine building. Even a complete emptying of the borated water storage tank (approximately 1,440,000 liters) onto this elevation would not provide the required water volume. Also, pipe breaks of the sizes postulated for this analysis would require hours of unmitigated flooding before they could even approach the required volume of water. Because of these considerations, flooding of both buildings at this elevation to a sufficient water depth is not deemed feasible. A flood in the turbine building would still leave several automatic safety systems available and a flood in the auxiliary building would leave the PCS available. Consequently, this level is not analyzed further.

3. CONCLUSIONS AND INSIGHTS

No areas were identified at Arkansas Nuclear One which are vulnerable to internal-flood-induced core damage accidents with a probability of greater than 1.0 E-06/Rx yr. Much of the active safe shutdown equipment is located on the upper levels which are very large. Electrical distribution equipment is located in individual areas which have significant drainage paths available in the immediate vicinity of the areas. Finally, the control room is protected by its location, its continuous manning, and drain paths immediately outside the area.

Specific plant and uncertainty insights are presented below:

Plant Insights:

No plant insights were developed for Arkansas Nuclear One since no specific areas were identified as being vulnerable to internal flooding.

Uncertainty Insights:

Some of the potential major areas for uncertainty are as follows:

- a) The potential for spray sources and/or submergence to actually cause damage to equipment.
- b) The effects of humidity from an internal flood on safety-related equipment.
- c) The method of partitioning used to determine generic room or building internal flood frequency from the internal flood occurrence data base. Similarly, the applicability of generic internal flood frequency data to an individual plant. Also included in the uncertainty of internal flood frequencies is the uncertainty associated with the size of the source of the internal flood.
- d) The human factors uncertainty associated with whether the operators will correctly isolate the flood source.
- e) The uncertainty associated with the performance of doors, dampers, and penetrations during an internal flood.

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APPENDIX F

EXTERNAL FLOOD ANALYSIS FOR THE
ANO-1 NUCLEAR POWER PLANT

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1.0 INTRODUCTION

1.1 Background

Sandia National Laboratories (SNL) is participating in the Task Action Plan (TAP) A-45 program to evaluate the adequacy of current licensing requirements for shutdown decay heat removal systems. As part of the SNL study, the response of decay heat removal systems to internal and external initiating events is being considered. The full scope of the analysis plan is reported in Reference 1.

Science Applications International Corporation (SAIC) is assisting SNL in the internal event and external event analyses being conducted for the U.S. Nuclear Regulatory Commission (USNRC). Jack R. Benjamin and Associates, Inc. (JBA) is a subcontractor to SAIC and provided the flood hazard and fragility data used in the TAP A-45 risk analyses. The external flood analyses for Arkansas Nuclear One-Unit 1 (ANO-1) are given in this report.

1.2 External Flood Risk Analysis

As stated in Reference 1, it is beyond the scope of the TAP A-45 program to perform an in-depth probabilistic risk assessment. Instead, the objective is to conduct an analysis that quantifies the significant threats to a plant. The analysis procedure used to perform the flood analysis embodies the basic philosophy of full-scope probabilistic risk assessment.

To evaluate the frequency of plant damage due to external flood events, the following tasks are performed: 1) plant familiarization, 2) hazard analysis, 3) fragility/vulnerability evaluation, 4) systems analysis, and 5) risk quantification. Figure 1-1 shows the relationship of these elements. The systems analysis and risk quantification were performed by SAIC and SNL, while the hazard and fragility analyses were conducted by JBA.

The steps in the analysis are summarized below.

- o Plant Familiarization - The purpose of this task is to gather information on the occurrence of external flood hazards at a plant and on the vulnerability of plant structures and equipment to external flooding. In assessing the reliability of shutdown decay heat removal systems, a review of available documentation (i.e., FSAR, plant drawings and topical reports) and a walkdown of the plant are performed.
- o Hazard Analysis - In the case of external flooding, the hazard analysis is conducted in two steps. First, a screening of the potential sources of flooding (i.e., upstream dams, local precipitation, storm surge, etc.) in the vicinity of the plant is made. For sources of flooding that can impact the plant site, an analysis is conducted to evaluate the frequency of occurrence of increasing levels of flood intensity (i.e., depth of inundation, hydrodynamic forces, etc.). An uncertainty analysis is also performed to assess the modeling or professional uncertainty in estimating the frequency of occurrence.
- o Fragility Evaluation - For structures and equipment items vulnerable to flood hazards, the fraction of failure defined as a function of a flood hazard parameter (i.e., depth of inundation), called a fragility curve, is assessed. The modeling uncertainty in the evaluation is also considered.
- o Systems Analysis - The shutdown decay heat removal systems analysis is performed by developing event trees and fault trees with external flooding as the initiating event. The component fragilities are used to evaluate the conditional fraction of failure of different safety systems.
- o Risk Quantification - The quantification of the likelihood of plant damage is assessed by properly weighting the conditional fraction of plant damage (i.e., system fragility curve) defined as a function of flood intensity, by the frequency of occurrence of each flood level.

By propagating the uncertainty in the flood hazard frequency assessment and the component fragilities, the probability distribution on the frequency of core melt is evaluated.

The scope of this report includes all of the above tasks.

1.3 Analysis Approach

To evaluate the contribution to core melt of external flood hazards, a simplified flood risk assessment approach was used. The flood risk analysis performed as part of TAP A-45 relied heavily on available information on regional and local flooding and any studies performed specifically for the plant. A screening procedure was used to eliminate sources of flooding as potential risk contributors where conservative arguments indicated this was reasonable. Within the scope of the TAP A-45 program, simplified procedures were then used to evaluate the frequency of occurrence of flood hazards that could impact the plant.

A conservative approach was used in developing the capacities of the structures and equipment to resist the effects of flooding. This procedure was based on the a prior/ assumption that the individual contribution of each of the components to core melt was not significant. When it was found in the systems analysis that external flooding was a significant contributor relative to other external and internal event initiators, then a more detailed look at components which dominate the flood contribution was required. By proceeding in this manner the analysis effort focused on the most significant contributors, and components which were not contributors were screened out by a defensibly conservative approach that required a minimum of effort.

1.4 Report Organization

In Section 2.0 the plant and site characteristics of ANO-1 are described from the perspective of the affects that external flood events have on shutdown decay heat removal systems. The plant site is described and the structure and equipment items vulnerable to external flooding are given.

In Section 3.0 the assessment of external flood hazards is discussed. Based on an initial screening of the possible sources of flooding, an assessment of the frequency of flooding at the plant site is made.

In Section 4.0 the results of a fragility evaluation of structures and equipment exposed to on-site flooding are reported.

In Section 5.0 the modeling of the plant's systems responses is performed.

In Section 6.0 the risk quantification is performed to determine the contribution to core melt frequency due to external flooding.

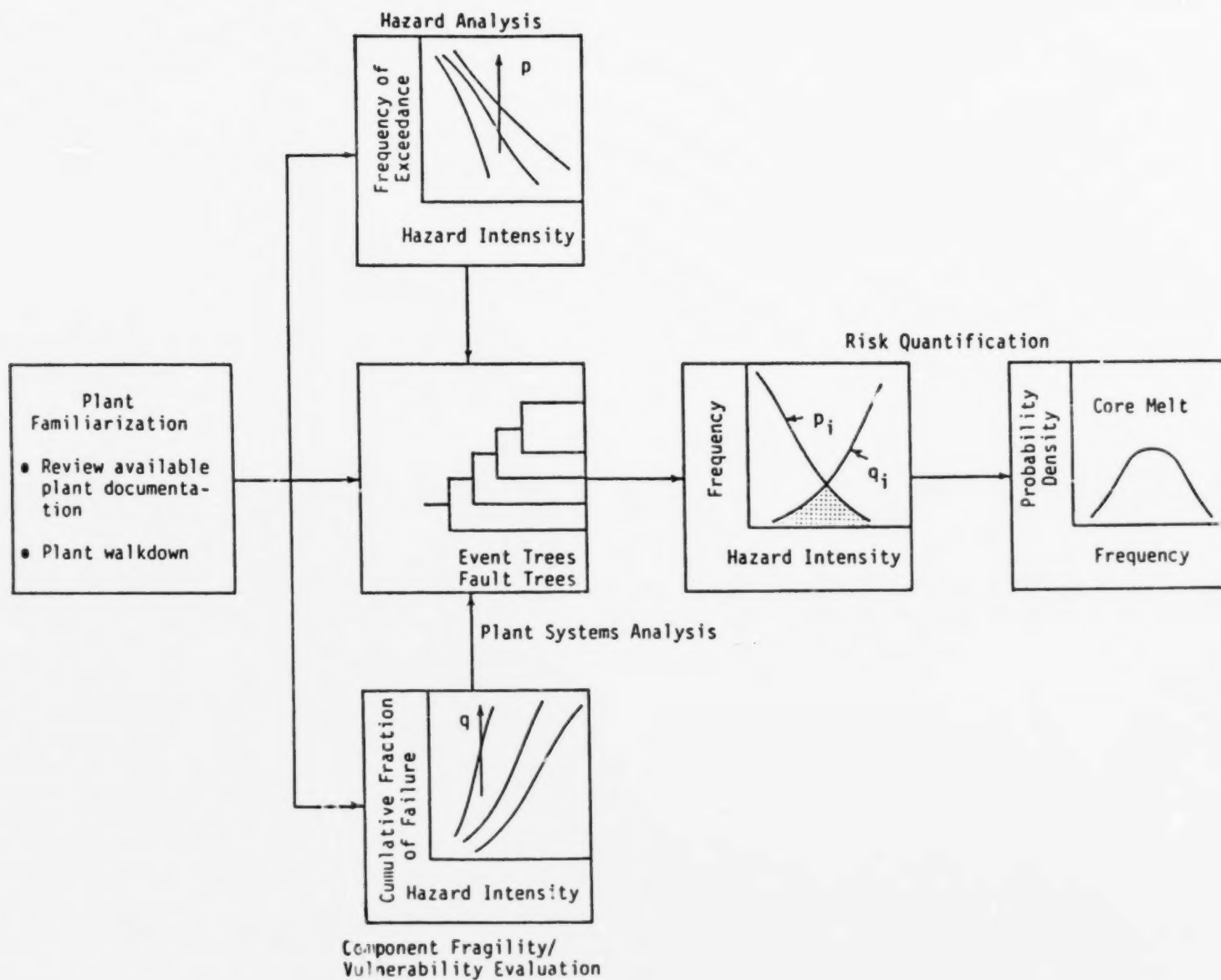


FIGURE 1-1 RISK ANALYSIS FLOWCHART

2.0 PLANT AND SITE CHARACTERISTICS

In this section a description of the ANO-1 plant and site characteristics is given. A plant walkdown of ANO-1 was conducted to gain familiarity with the plant and site characteristics and to determine the vulnerability to external flood hazards of structures and equipment important to decay heat removal. Plant design documents, including the Final Safety Analysis Report (FSAR) (Refs. 2, 3) and construction drawings were obtained and reviewed. Discussions with plant and utility personnel were conducted to obtain an understanding of the protection provided to resist the effects of flooding.

In Section 2.1 the plant location is discussed from the perspective of the external flood analysis, including a general description of the flood hazards at ANO-1. In Section 2.2 a discussion of the plant characteristics with respect to external flooding is given. A summary of the structures and equipment vulnerable to flood hazards is provided in Section 2.3.

2.1 Plant Location and Flood Hazard

ANO-1 is located on a peninsula on the east bank of Dardanelle Reservoir on the Arkansas River, approximately six miles west-northwest of Russellville, Arkansas. The plant site is at navigation mile 212 (river mile 264.3), located upstream of Dardanelle Lock and Dam (L&D) No. 10 which maintains a normal reservoir elevation of 338 feet National Geodetic Vertical Datum (NGVD). Approximately 50 miles upstream of ANO-1 is Ozark L&D No. 12. These dams are part of the McClellan-Kerr Arkansas River Navigation System (Fig. 2-1). Plant grade at the site is 353 feet NGVD. The ground-floor elevation of plant structures is 354 feet NGVD.

The topographic relief of the area surrounding the plant is sloping gently from the hills (about 500 feet NGVD) toward two major coves on the west bank of the Dardanelle Reservoir and a major cove of the Illinois

Bayou on the east bank. Figure 2-2 shows the topography of the plant site and the surrounding area.

The principal cause of flooding in the area is the Arkansas River and its tributaries. Historically, the worst floods have occurred in the spring. In addition, intense rainfall can also cause local flooding.

The largest recorded flood on the Arkansas River near ANO-1 occurred in May, 1943. The peak discharge at Dardanelle, Arkansas (2 miles downstream from the dam) was 683,000 cubic feet per second (cfs). The flood reached a peak stage of approximately 343.5 feet NGVD which was 9.5 feet lower than plant grade of 353 feet NGVD. Table 2-1 summarizes the major floods on the Arkansas River at Dardanelle, Arkansas. According to the ANO-1 Unit 1 FSAR (Ref. 2), daily streamflow records for the period January 1923 to September 1951 for the Dardanelle gauging station have been adjusted by the Corps of Engineers to reproduce flows as they would have been regulated by the complete system of dams upstream. The maximum regulated daily discharge during this period would have been 480,000 cfs (approximately 341 feet NGVD at ANO-1). Dardanelle Dam is designed to discharge up to 900,000 cfs without exceeding a maximum water level of 338 feet NGVD.

The Arkansas River Navigation System consists of 17 locks and dams. Thirteen of these are simple locks and dams providing navigation lifts of 30 feet or less. Dardanelle, Ozark, Robert S. Kerr, and Webber Falls Dams are higher, with lifts up to 54 feet and include some storage for hydropower generation (see Fig. 2-1). Table 2-2 lists these structures and their location. According to Reference 2, only Ozark Dam presents a potential threat to ANO-1.

2.2 Plant Description and Flood Design Basis

Figure 2-3 shows a view of the general arrangement of plant structures and outside equipment items located outside at ANO-1. As part of the original design, plant structures and components have been constructed to withstand the effect of extreme flood scenarios including

local intense precipitation, upstream dam failure and the probable maximum flood (PMF) (Ref. 2,3).

The flood level produced at the plant site due to the PMF is 358 feet. In the event that Ozark Dam fails during the PMF, a maximum water surface elevation of 361 feet, 8 feet above plant grade would result. The effect of wind generated waves was also considered in the plant design. According to the FSAR (Ref. 2) it was concluded that wind generated waves do not constitute a significant hazard to the plant. The design basis flood elevation was therefore defined to be 361 feet.

Safety-related equipment which is part of the shutdown decay heat removal system is located in the Reactor Building, Auxiliary Building, Turbine Building, and Intake Structure. Outside are the Condensate Storage Tanks (CST), Borated Water Storage Tanks (BWST), and Emergency Diesel Fuel Oil Storage Tank (Fig. 2-3).

Category I plant structures (Reactor Building, Auxiliary Building, portion of Intake Structure housing, Emergency Diesel Fuel Storage Vault) have been evaluated to assess the effects of flood level elevation to plant grade.

Safety-related equipment is either located on floors at or above elevations 369, or in Category I structures with wall thicknesses of at least 2 feet. Watertight doors are provided at entrances to the Auxiliary Building such that flood protection is provided to at least elevation 361 feet.

In the event that a major flood occurs, emergency procedures (Ref. 4) exist to shutdown the plant and to secure vulnerable areas. The procedures call for establishing contact with the Corps of Engineers to obtain accurate peak flood forecasts and updates should the lake level exceed elevation 340 feet.

According to the FSAR (Ref. 2) a flood of a magnitude of the PMF will be forecast about five days prior to arrival at the plant site. The

plant is expected to be shutdown by the time the flood level reaches 354 feet.

2.3 Vulnerable Structures and Equipment

Based on a review of plant documents and a walkdown of the plant, an assessment of the location and vulnerability of safety-related equipment was made. Table 2-3 summarizes this information.

The switchyard is situated east of the plant, approximately at elevation of 354 feet. Thus for floods that could impact the plant, failure of the switchyard and loss of offsite power is anticipated.

TABLE 2-1
MAJOR HISTORIC FLOODS AT DARDANELLE
(Refs. 2, 5)

Year	Peak Discharge (cfs)
1938	396,000*
1943	683,000*(480,000)**
1957	471,000
1973	380,000
1982	326,900

* unregulated flow

**equivalent regulated flow

TABLE 2-2
ARKANSAS RIVER DAMS UPSTREAM OF ANO-1

Name	Location	Navigation Mile*
Dardanelle L&D (No. 10)	Dardanelle, Arkansas	205.5
Ozark L&D (No. 12)	Ozark, Arkansas	256.8
Robert S. Kerr L&D (No. 15)	Oklahoma	336.2
Webber Falls L&D (No. 16)	Webber, Oklahoma	368.9

* ANO-1 is located at navigation mile 212.

TABLE 2-3
VULNERABLE STRUCTURES AND EQUIPMENT

Structures/Equipment	Location	Vulnerability
480V Switchgear	Located on the east side of the Auxiliary Building, elevation 372 feet NGVD.	The 480V switchgear are vulnerable if flood elevations exceed 372 feet NGVD.
Emergency Switchgear Room	Located on the east side of the Auxiliary Building, elevation 372 feet NGVD.	The 4.16kV switchgear are vulnerable if flood elevations exceed 372 feet NGVD.
Emergency Feedwater Pumps	Located on the east side of the Auxiliary Building, elevation 335 feet NGVD.	Emergency Feedwater Pumps are vulnerable if the Auxiliary Building is flooded. Flood protection is provided to elevation 361 feet NGVD. Vulnerable if the exterior walls of the Auxiliary Building fail.
High Pressure Injection System (HPIS)	HPIS pumps are located on the north side of the Auxiliary Building, elevation 335 feet NGVD.	HPIS pumps are vulnerable if the Auxiliary Building is flooded. Flood protection is provided to elevation 361 feet NGVD. Vulnerable if the exterior walls of the Auxiliary Building fail.
Low Pressure Injection System (LPIS)	LPIS pumps are located on the northwest side of the Auxiliary Building, elevation 317 feet NGVD.	LPIS pumps are vulnerable if the Auxiliary Building is flooded. Flood protection is provided to elevation 361 feet NGVD. Vulnerable if the exterior walls of the Auxiliary Building fail.
Diesel Generators	Located on the northwest side of the Auxiliary Building, elevation 369 feet NGVD.	Vulnerable if flood elevation is above 369 feet NGVD.

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TABLE 2-3

VULNERABLE STRUCTURES AND EQUIPMENT (CONTINUED)

Structures/Equipment	Location	Vulnerability
Battery Room	Located on the north side of the Auxiliary Building, elevation 372 feet NGVD.	Vulnerable if flood elevation is above 372 feet NGVD.
Cable Spreading Room	Located on the northeast side of the Auxiliary Building, elevation 374 feet NGVD.	Vulnerable if flood causes structural failure or leakage.
Control Room	Located on the north side of the Auxiliary Building, elevation 386 feet NGVD.	Vulnerable if flood causes structural failure or leakage.
Reactor Building Spray System (RBSS)	RBSS pumps are on the northwest side of the Auxiliary Building, elevation 317 feet NGVD.	Vulnerable if Auxiliary Building is flooded. Flood protection provided to elevation 361 feet NGVD. Vulnerable if the exterior walls of the Auxiliary Building fail.
Service Water Pumps	Located in the Intake Structure, elevation 366 feet NGVD.	Flood protection is provided to 361 feet NGVD.
Switchyard	Located east of the Turbine Building, elevation 354 feet NGVD.	Vulnerable if flood level exceeds elevation 354 feet NGVD.



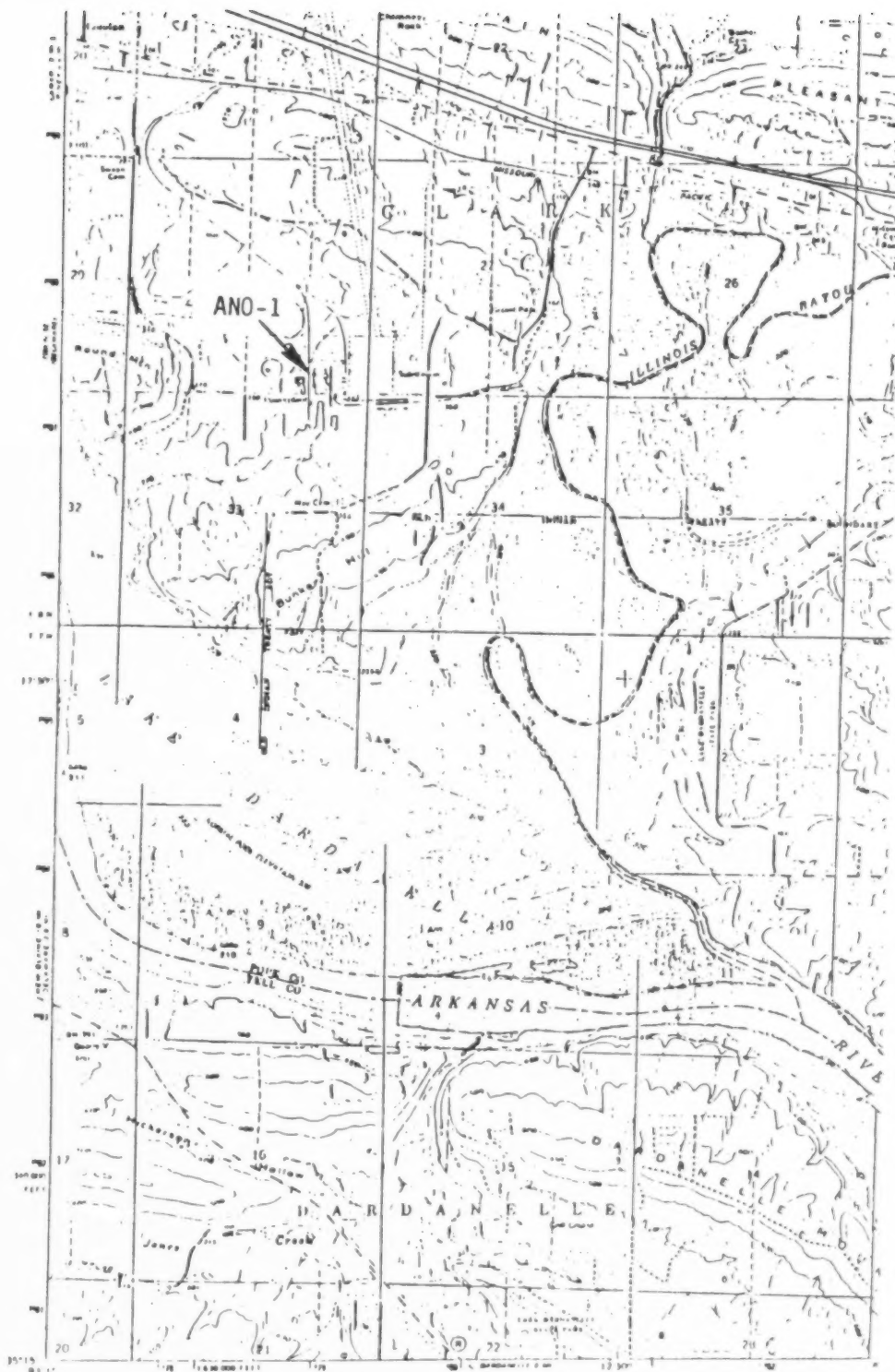


FIGURE 2-2 ANO-1 AND SURROUNDING AREA
(From USGS Topographic Map)

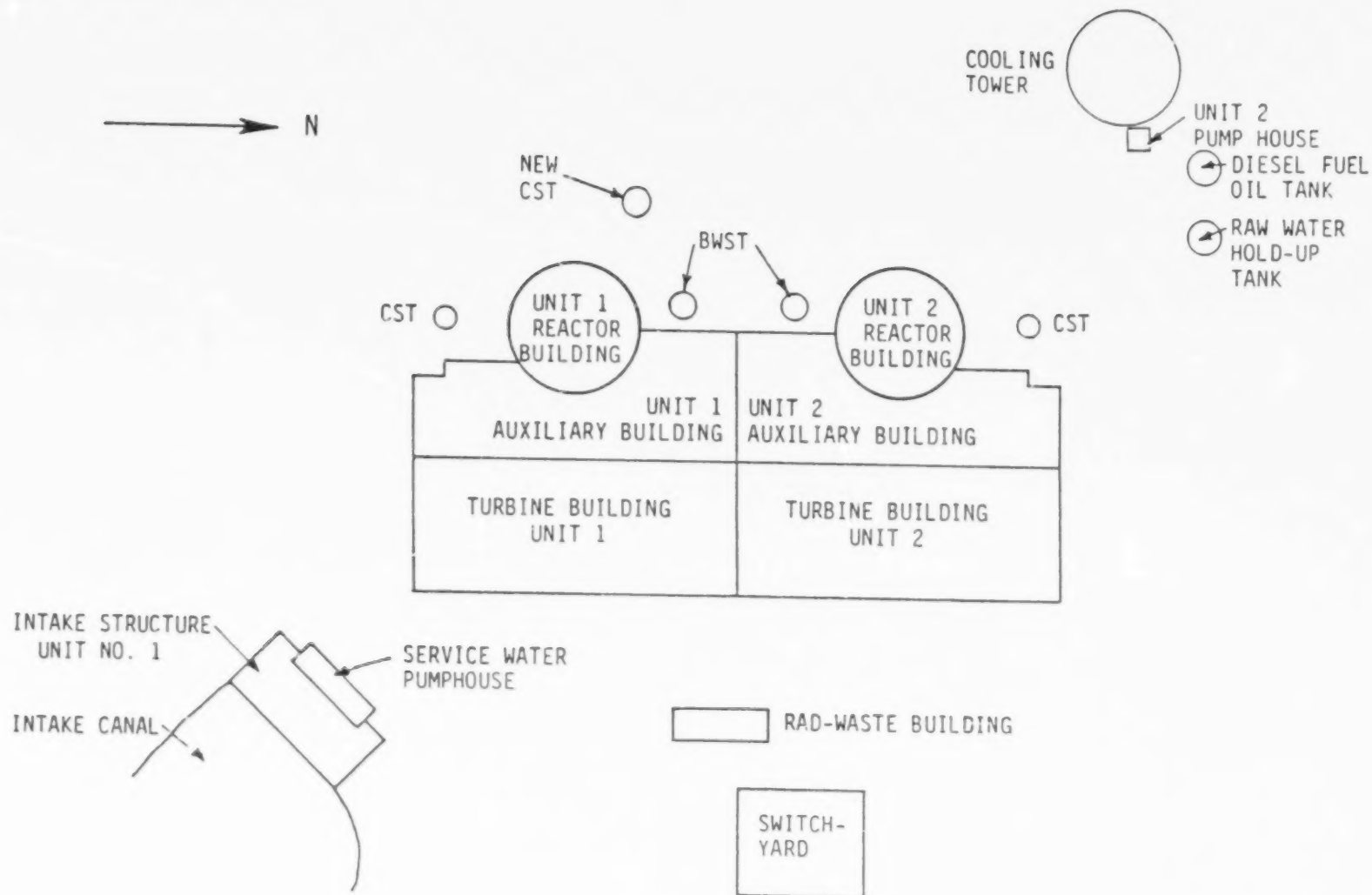


FIGURE 2-3 PLANT GENERAL ARRANGEMENT

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3.0 FLOOD HAZARD ANALYSIS

Unlike other natural hazards such as earthquakes and extreme winds, external flood events may not present an adverse hazard to safety-related structures or equipment at a particular plant. Due to topographic or other physical conditions, flooding at a site may be precluded from occurring at all, or it may have an extremely low frequency of occurrence. This is due in part to the fact that the design basis floods for nuclear plants are generally conservative. However, since the uncertainty in design floods was not considered, the likelihood of extreme flooding and the contribution to plant risk is not known.

In this flood hazard assessment a series of steps were taken to evaluate the significant hazards that could impact a plant. In the first step the potential sources of flooding in the vicinity of the plant were identified. Next, a screening analysis was performed to establish for each source of flooding whether the plant site could be impacted. The results of the screening analysis for ANO-1 are described in Section 3.1.

For flood hazards which could adversely impact the plant, a simplified analysis was conducted to evaluate the probability distribution on the frequency of flooding the site. The hazard analysis methodology is described in Section 3.2. The results of the analysis are given in Section 3.3.

3.1 Screening Analysis

In the screening analysis potential sources of flooding at ANO-1 were identified and evaluated. The objective of this first phase of the flood hazard assessment was to establish which flood events required direct consideration in the risk analysis. For those sources of flooding that do not adversely impact the plant, the basis for eliminating them from further consideration is provided.

The scope of the screening analysis generally involved simplified, conservative assessments to determine whether a particular source of flooding adversely impacted plant structures and equipment. If on this basis the integrity of plant components necessary for the removal of decay heat was not jeopardized, no further consideration was warranted.

Table 3-1 lists the potential sources of flooding in the vicinity of ANO-1. These sources of flooding were identified based on information provided in the ANO-1 FSAR (Refs. 2, 3), and a review of information provided in References 4 through 12.

Arkansas River Flooding

The Arkansas River is a source of potential flooding in the area near ANO-1. Historically, large floods that have occurred are the result of storm runoff. The largest flow at the plant site occurred in May 1943 with a peak discharge of 683,000 cfs. This flood produced a maximum elevation of about 343.5 feet NGVD, which was 9.5 feet lower than plant grade of 353 feet. Table 2-1 summarized the flood events that have occurred with a peak discharge greater than 300,000 cfs.

The design basis flood for ANO-1 considers the effect of Probable Maximum Flood conditions on the Arkansas River and coincident failure of Ozark Dam. According to Reference 2, the Corps of Engineers estimate the PMF flow at Dardanelle Dam as 1,500,000 cfs. This flow results in a flood stage elevation of 358 feet at the plant site, on a straight line variation between Dardanelle Dam (353 feet) and Ozark Dam (389.5 feet).

For purposes of the screening analysis, the frequency of an event this size or greater was estimated. To do this, the results of a frequency analysis for the Arkansas River at navigation mile 212, obtained from the Corps of Engineers (Table 3-1), was conservatively extrapolated to obtain an order of magnitude assessment of the frequency of exceeding 1,500,000 cfs. The frequency of exceedance was estimated

between $1.0 \cdot 5$ and $1.0 \cdot 6^1$. This frequency was high enough that flooding of the plant site from the Arkansas River was incorporated in the hazard analysis.

Upstream Dam Failures

The Arkansas River navigation system consists of 17 locks and dams (Fig. 3-1). Thirteen of these are simple lock and dam systems that provide navigation lifts of 30 feet or less. Dardanelle, Ozark, Robert S. Kerr, and Webber Falls Dams are higher with lifts up to 54 feet. Of these four dams, Dardanelle is located downstream of ANO-1. In Reference 13, the U.S. Nuclear Regulatory Commission published an inventory of dams that could impact nuclear power plants, should they fail. A review of the inventory indicated that dams upstream of ANO-1 do not impact the plant. However, failure of Ozark Dam (the closest upstream dam to ANO-1) during probable maximum flood conditions could flood the plant site to elevation 361 feet (Refs. 2, 3).

An analysis of the failure of dams upstream of Ozark Dam has been performed in Reference 3. Should failure of any of these dams occur, the additional discharge that would result can be safely conveyed by the spillway at Ozark Dam. Downstream at ANO-1, the resulting flow does not exceed plant grade.

The failure of Ozark Dam which is a concrete, gravity structure, was analyzed in the ANO-1 FSAR (Ref. 2). Assuming complete and instantaneous failure of the dam at maximum pool conditions, it was conservatively estimated that the maximum water elevation at the plant site would be 342.6 feet (10.4 feet below plant grade). The peak of the flood would arrive about 15 hours after the dam failed. Therefore random failure of Ozark Dam due to any initiating (excluding extreme floods) is not hazard.

¹ $1.0 \cdot 6 = 1.0 \times 10^{-6}$

Failure of Ozark Dam under PMF conditions was also investigated. Incorporating the surcharge storage from Robert S. Kerr Dam that would contribute to the flow, the head differential at Ozark Dam was estimated at 11.5 feet. According to References 2 and 3, failure of Ozark Dam at the peak of the flood produces a maximum elevation of 361 feet at the plant site. The peak of the flood would arrive approximately 18 hours after the dam failed.

In no case does failure of Ozark or any other dam upstream during normal (non-flood) conditions, pose a flood hazard to ANO-1. Therefore, dam failure is not considered in the flood hazard assessment. However, since coincident failure of Ozark Dam during the PMF contributes to the flood hazard, dam failure is considered in the flood analysis.

Local Precipitation

In some cases, special problems can arise due to local intense rainfall. These could include ponding on site, leakage into plant structures, and collection of water, snow and ice on building roofs. As part of the screening analysis these areas were investigated.

Climatological data for the plant area is available for Fort Smith, Arkansas, Little Rock, Arkansas and Ozark, Arkansas (Refs. 6, 7 and 8). The average annual rainfall for the area is approximately 50 inches. The precipitation is quite uniform through the year. The maximum monthly and daily rainfall recorded at either Fort Smith, Ozark, or Little Rock was 18.04 inches in January 1937 and 9.58 inches in April 1913. Based on an analysis of rainfall data, the National Weather Service has estimated the 24-hour probable maximum precipitation (PMP) to be approximately 39 inches (Ref. 9). The mean annual total snowfall in the region is approximately 5 inches. The maximum 24-hour accumulated snow was 13 inches in January 1983.

Detailed information about the site drainage was not available, however a summary of the design/analysis results are reported in the FSAR

(Refs. 2, 3). Information on the original plant drainage system indicates it was designed for a 10-year recurrence frequency storm and checked for a 100-year storm based on information provided in Reference 3. It was concluded that localized ponding could occur on-site in the vicinity of plant structures. The occurrence of on-site ponding is expected to be limited (Refs. 2, 3). In any event ponding on-site is considered to be a self-limiting process for a number of reasons. First, the topography in the vicinity of the plant slopes toward Dardinelle Reservoir and the crest of local roadways is below the ground floor level of safety-related structures. As a result, the extent of ponding is limited and runoff to the reservoir is expected. Also, the ground floor of plant structures is at least six inches higher than the ground surface and there are no openings where water could enter.

Secondly, flood protection is provided to elevation 361 feet NGVD. Local ponding to this elevation is not a realistic scenario, given the first item listed above.

In addition to ponding on-site, accumulation can occur on the roof of structures with parapet walls and poor drainage. The structures which could be adversely impacted by collection of water are the Auxiliary Building and the Intake Structure. In the case of the Auxiliary Building, all safety-related equipment is located in a reinforced concrete structure with roof slab thicknesses of 2 to 3 feet. The reinforced concrete roof of the Intake Structure is 2 feet thick. In the event roof drains become clogged (partially or fully), water would accumulate.

The spent fuel pool is located at elevation 404 feet NGVD. The upper level of the Auxiliary Building is housed in a steel frame superstructure with metal siding and roof decking. The parapet walls on this structure are approximately 4 feet high. Should accumulation result in roof damage, leakage into the building could occur.

Based on design drawings and roof drain specifications, the drainage capacity of the Auxiliary Building and Intake Structure was estimated conservatively to be 8 inches/hour. Rainfall atlas maps published by the National Weather Service were used to determine the frequency of exceedance of rainfall by fitting a Log-Pearson Type III distribution to the frequency information provided in (References 10, 11 and 12). For a one-hour duration rainfall the frequency of exceeding 8 inches was about 1.0-5 per year. If the drainage outlets are blocked such that they lose one-half their capacity, approximately 4 inches of water would accumulate, which is approximately equivalent to the design basis roof load of 20 psf. The frequency of exceeding a cumulative rainfall of 8 inches in storm durations ranging from one hour to two days is 0.18 per year.

There is no readily available data to assess the likelihood that the storm drains will be fully or partially clogged. If it is assumed that the roof decking has a design factor of safety of 2.0, and the probability of all roof drains being clogged is 1.0, then the frequency of failure is conservatively estimated to be 1.0-5. Detailed information about the roof decking design could not be obtained, however a factor of safety of 2 in the is believed to be conservative.

Longer duration storms can occur. However, the rainfall intensity per hour would generally be less than half the roof drainage capacity. The probable maximum precipitation (PMP) for a 24-hour storm is 39 inches (Ref. 9). If 100% of this rainfall accumulated on the roof it would exceed the capacity of the roof deck system. However, for the distribution of rainfall intensity as shown in Table 3-3, it is expected to be less than the drainage capacity of the roofs. The frequency of the 24-hour PMP was estimated to be less than 1.0-7 per year.

Based on available information, if the roof drains of the Auxiliary Building perform adequately, the frequency of exceeding the drainage capacity and water accumulating on the roof is small, less than 1.0-5 per year. If however, roof drains do become clogged, accumulation will

occur. It is estimated that accumulation of greater than 8 inches must occur to reach the capacity of the roof decking. The fact that no information on the design strength of the roof decking could be obtained, limits the capability to assess the likelihood of damage. Potential problems with water accumulation on the Auxiliary Building or Intake Structure is easily rectified by adding scuppers that limit the amount of water that can be stored. Scuppers were used in the design of ANO-1 Unit 2 roof systems.

Icing of the Arkansas River

With respect to ice flooding, the FSAR indicates ice formation is light and does not cause significant flooding problems (Ref. 3). On this basis it was concluded that a detailed hazard analysis for ice flooding is not required.

Wave Action

According to the FSAR, wave action could accompany a large flood on the Arkansas River. Based on the calculations reported in Reference 2, waves that occur on site would have little impact on plant structures. The principal effects of wave action would be spray and runup on the walls of reinforced concrete structures. These effects are not considered a significant hazard.

3.2 Hazard Assessment

Based on the results of the screening analysis, a hazard assessment for the effects of flooding on the Arkansas River was recommended. From the screening analysis it was concluded that the frequency of floods with the potential to impact safety-related structures and equipment was high enough such that a probabilistic hazard analysis be performed.

The possibility of flooding the ANO-1 site was considered in the plant design basis. To prevent damage to the reactor, emergency operation procedures were developed to shutdown and maintain the plant in a safe condition (Ref. 4). Should the lake level reach elevation 340 feet NGVD (13 feet below plant grade) and is continuing to rise, plant shutdown procedures are initiated. However, no specification is given regarding the time at which plant shutdown should occur.

3.2.1 Methodology

Frequency of Flooding

There is considerable difference of opinion regarding which, if any, procedure should be used to evaluate the frequency of extreme floods. Clearly, no single statistical model or probabilistic methodology has received wide acceptance. Furthermore, there is little experience in the probabilistic assessment of extreme flood hazard phenomena the size of which impact nuclear stations since they are generally designed to conservative standards. In the flood hazard analysis for TAP A-45, a simplified approach was used that produced estimates of the frequency of floods that impact the plant and assessed the uncertainty in the frequency distribution.

There are two general approaches available to evaluate the frequency of flooding on the Arkansas River. The first, more traditional approach, is to fit a statistical frequency distribution to stream gauge data gathered at or near the site of interest. In the U.S., typically 30 to 60 years of stream gauge data or less are available. In the second approach, analytical methods are used to model the response of a basin to storms of varying size or intensity. With a mathematical model of the physical process as input, a probabilistic model is developed to evaluate the frequency of occurrence and magnitude of floods due to rainfall or snowmelt, or a combination of these. This approach involves storm and basin data collection, extensive analysis of regional storm data, basin modeling and computer analysis. Although preferred, this approach is

beyond the scope of TAP A-45. Therefore, a statistical frequency distribution approach was used to evaluate the frequency of flooding on the Arkansas River.

To evaluate the frequency of flooding at ANO-1, two steps were taken. First, a frequency analysis of the river gauge data on peak discharge at Dardanelle, Arkansas was analyzed. Second, based on a stage-discharge relation for the ANO-1 section of the river, the frequency of exceedance of peak flood elevation was determined. These steps are shown in Figure 3-2. Multiple curves are shown to reflect the uncertainty in each step in the analysis.

In the flood hazard analysis the following sources of uncertainty were considered:

- choice of the statistical frequency distribution
- statistical uncertainty about a given frequency model
- stage-discharge relationship

For each part of the analysis, alternative hypotheses were considered to characterize the uncertainty. Subjective probability weights were assigned to each alternative hypothesis to express the degree-of-belief that it represents the true alternative. For example, three alternative statistical frequency distributions could be selected to characterize the uncertainty in defining the mean frequency of exceedance. Subjective probability weights would then be assigned to each model, expressing the credibility given to it.

To estimate the frequency distribution on peak discharge, the following statistical models were selected (Ref. 14):

- Lognormal
- Log-Pearson Type III
- Gumbel or Extreme Value Type I

Based on the fit of each statistical model to the data, a probability distribution on the model parameters was obtained. In the case of the Lognormal model, for example, a joint normal distribution on the mean and standard deviation of the logarithms of the flow data was obtained, $f(m, \sigma)$. By discretizing the sample space of m and σ values, a family of hazard curves was derived for the lognormal model that expresses the statistical variability in the estimate of the frequency of exceedance. The probability weight assigned to each curve, r_i , is equal to,

$$r_i = \int_m^{m+\Delta m} \int_{\sigma}^{\sigma+\Delta \sigma} f(m, \sigma) d\sigma dm \quad (3-1)$$

This theme was repeated for the other models to determine their statistical variability.

In the ANO-1 FSAR (Refs. 2, 3) no stage-discharge relationship was estimated. However, the flood frequency study conducted by the Corps of Engineers (COE) provides an estimate of flood stages due to river flooding (Ref. 5). These values are represented in Table 3-2. In addition the COE study gives the flood stage for the standard project flood (SPF). In addition, the FSARs for ANO-1 (Refs. 2, 3) give an estimate of the flood stage corresponding to the PMF as well as the flow required to overtop the levees along the Arkansas River (elevation 346 feet NGVD). These values were used as a basis to derive a stage-discharge relationship at ANO-1 (Fig. 3-3).

To characterize the uncertainty in predicting flood stages at ANO-1, a family of three stage-discharge curves was developed. Figure 3-3 shows the stage-discharge relationships and the probability weights assigned to each curve. The variation in the prediction of flood stage was estimated based on engineering judgment, the differences in the two curves reported in the FSAR for high and low flow conditions, and estimates of modeling uncertainty for flood mapping (Ref. 15). The probability weights were assigned subjectively.

By taking all possible combinations of the alternative frequency-discharge curves and stage-discharge relations, a family of flood hazard curves was developed (Fig. 3-5). The probability weight assigned to each curve is simply the product of the probabilities assigned to each hypothesis included in a particular combination. Figure 3-4 shows part of a logic diagram that illustrates the uncertainty analysis. The product of the analysis is a probability distribution on the frequency of flooding, composed of individual elements, $v(S)_\ell$ and $(p_i q_j r_k)$, where $v(S)_\ell$ is the ℓ th estimate of the frequency distribution on flood stage, S , and $(p_i q_j r_k)$ is the subjective probability that it is the "true" hazard curve. The product of this analysis expresses the uncertainty in estimating the frequency of exceedance. From the resultant probability distribution on frequency, a mean flood hazard curve was determined and used to estimate the mean frequency of core melt.

Flood Warning Time

According to the ANO-1 Natural Emergency Procedures (Ref. 4), the plant will shutdown in the event it appears a major flood could occur. The procedures are defined according to observed river levels. Ultimately, if the river reaches elevation 340 feet, 13 feet below plant grade, and is still rising, emergency procedures such as verification that the emergency diesel fuel tanks are full, that tank inlets are isolated, that vault sump pumps are in "auto" and that water-tight doors are secured, are instituted. However the natural emergency procedures do not specifically call for the plant to be shutdown.

3.2.2 Results

Flood Frequency

The quantification of the flood hazard at ANO-1 was performed using the steps outlined in Figure 3-4. For each of the three statistical frequency distributions, a family of 5 hazard curves that expresses the statistical uncertainty in the estimate were determined. In all 15

hazard curves (3 models x 5 curves per model) that define the frequency of exceedance of peak discharge were estimated.

Two sources of information were available to estimate the frequency of flooding on the Arkansas River. The first source of input was river gauge data obtained at Dardanelle, approximately 5 miles downstream (Table 3-4). The second data source consisted of the results of a COE flood frequency study for the Arkansas River (Ref. 5). Both sources of information were used in the flood frequency analysis.

To perform a statistical analysis using the results of the flood frequency study performed by the COE, this information was used as data to determine the parameters of a given statistical distribution. Based on the parameters derived in the fitting process and knowledge of the length of record used in the original study, the statistical variability in the estimate of the distribution parameters can be determined. Once the parameters were determined, the results could be extrapolated to estimate the frequency of flooding plant structures.

Each data source was used to estimate the parameters of the three statistical distributions used in the study. To obtain final estimates of the parameters for each model, a Bayesian weighting procedure was used (Ref. 17). The procedure assigned relative weights in proportion to the length of record in each data set. The COE flood frequency study made use of 35 years of data, while the gauge data at Dardanelle were for a 21-year period. Therefore higher relative weight was assigned to the parameter estimates obtained using the COE flood frequency data.

Subjective probability weights were assigned to each statistical frequency distribution to reflect the belief that any one of them is the correct one. The following weights were assigned:

<u>Model</u>	<u>Subjective Probability</u>
Lognormal	0.33
Log-Pearson Type III	0.33
Gumbel	0.34

Equal weight was given to the three models since each is considered to be equally speculative in terms of its ability to predict the frequency of extreme floods on the Arkansas River

By an appropriate transformation using the stage-discharge relation in Figure 3-3, a family of hazard curves defining the frequency of exceeding flood levels at ANO-1 was determined.

In all 45 frequency curves on flood stage were estimated (15 frequency curves for peak discharge times 3 stage-discharge relations). The 45 curves were then synthesized into 5 curves of approximately equal weight. Figure 3-5 shows the resulting 5 hazard curves and the mean curve. Table 3-5 reports the frequency values for the 5 curves as well as the mean hazard curve.

Referring to the mean hazard curve in Table 3-5, the mean frequency of exceeding plant grade of 353 feet is approximately 3.2-5 per year. the frequency of exceeding the flood protection, which is provided to elevation 361 feet is approximately 3.6-6 per year.

Coincident Failure of Ozark Dam

In the event of an extreme flood, Ozark Dam could be overtopped, resulting in a breach of the dam. For PMF conditions, coincident failure of the dam adds 3 feet to the peak of the flood. No information is available to estimate the flood peak if dam failure occurs during floods smaller than the PMF. Conservatively, an estimate of the flood hazard due to the coincident occurrence of extreme floods and failure of Ozark Dam, an increase in the peak flood elevation of 3 feet could be added to the flood levels reported in Table 3-5. This approach assumes dam

failure occurs with probability 1.0, given the occurrence of a large flood and that the contribution of the dam breach to the peak flood level is 3 feet.

Based on this approach, the annual frequency of exceeding plant grade is 9.0-4. The frequency of exceeding minimum flood protection (e.g., elevation 361 feet NGVD) is 7.22-6.

Flood Warning

In order to implement the flood emergency procedures at ANO-1 adequate time must be available. According to the FSAR, the Corps of Engineers does monitor the water level on the Arkansas River during extreme floods. Past experience indicates that warning is available in the case of extreme flooding. No specific information on the rate of rise of the river during the PMF was available to calibrate the actual warning time that could be anticipated.

TABLE 3-1

SCREENING ANALYSIS SUMMARY OF EXTERNAL FLOOD HAZARDS
AT ANO-1 NUCLEAR STATION

Source of Flooding	Type of Flooding	Status
Arkansas River	Flooding on the river occurs due to storm runoff. Wave action can occur.	Hazard analysis required.
Failure of Ozark Lock and Dam No. 12	Ozark Dam is located 50 river miles upstream of the ANO-1 plant.	Failure of Ozark Dam during the PMF could flood the plant, but it is not a significant contributor to failure. Maximum height of water below essential equipment elevations.
Multiple Upstream Dam Failures	9 dams are located upstream on the Arkansas River.	No analysis required. Ozark Dam could pass the flow safely.
Local Precipitation	Intense rainfall can result in local runoff and collection of water on building roofs.	Collection of water on the Auxiliary Building roof or Intake Structure roof cannot be ruled out as a contributor to leakage into the Auxiliary Building failure. On-site flooding is possible, the amount of ponding is self limiting and it does not exceed the design basis.
Icing of the Arkansas River	Ice jams cause the river to back up.	Historically, not a significant cause of flooding on the Arkansas River; any contribution to flooding is assumed to be incorporated in the analysis of the river gauge data.

TABLE 3-2
DISCHARGE FREQUENCY ANALYSIS
OF NAVIGATION MILE 212, ARKANSAS NUCLEAR ONE SITE
(Ref. 5)

Flood Frequency (years)	Discharge (cfs)	Elevation (feet NGVD)
10	305,000	339.3
20	360,000	339.8
50	430,000	340.5
100	490,000	341.2

TABLE 3-3
PROBABLE MAXIMUM PRECIPITATION (PMP)
(Ref. 3)

Hour	Cumulative Precipitation (inches)
1	11.7
2	16.3
3	20.7
4	24.3
5	27.6
6	30.6
12	36.0
24	39.0

TABLE 3-4
DARDANELLE STATION DATA
(Ref. 16)

Year	Maximum Inflow (x 10 ³ cfs)
1964	123.6
1965	200.0
1966	90.0
1967	210.0
1968	195.0
1969	245.0
1970	165.0
1971	325.0
1972	380.0
1973	331.8
1974	256.1
1975	160.0
1976	228.0
1977	134.3
1978	122.5
1979	76.8
1980	58.2
1981	182.7
1982	327.0
1983	142.2
1984	241.4

TABLE 3-5
ARKANSAS RIVER FLOOD HAZARD SUMMARY

Curve	Subjective Probability	Peak Flood Elevation (feet NGVD)								
		340	344	348	352	354	356	358	360	362
1	0.192	1.29-2	2.76-5	3.16-7	4.73-9	8.69-10	1.85-10	4.07-11	1.24-11	2.45-12
2	0.185	2.68-2	1.74-4	4.32-6	1.46-7	3.60-8	1.32-8	3.51-9	1.20-9	3.88-10
3	0.228	4.02-2	5.41-4	2.65-5	1.53-6	4.73-7	1.84-7	6.80-8	3.16-8	1.39-8
4	0.190	5.82-2	1.41-3	1.20-4	1.36-5	5.72-6	2.61-6	1.21-6	6.60-7	3.63-7
5	0.205	9.75-2	5.79-3	9.78-4	2.03-4	1.05-4	5.89-5	3.40-5	2.13-5	1.34-5
	Mean Hazard	4.76-2	1.62-3	2.30-4	4.46-5	2.27-5	1.26-5	7.22-6	4.50-6	2.82-6

F-35

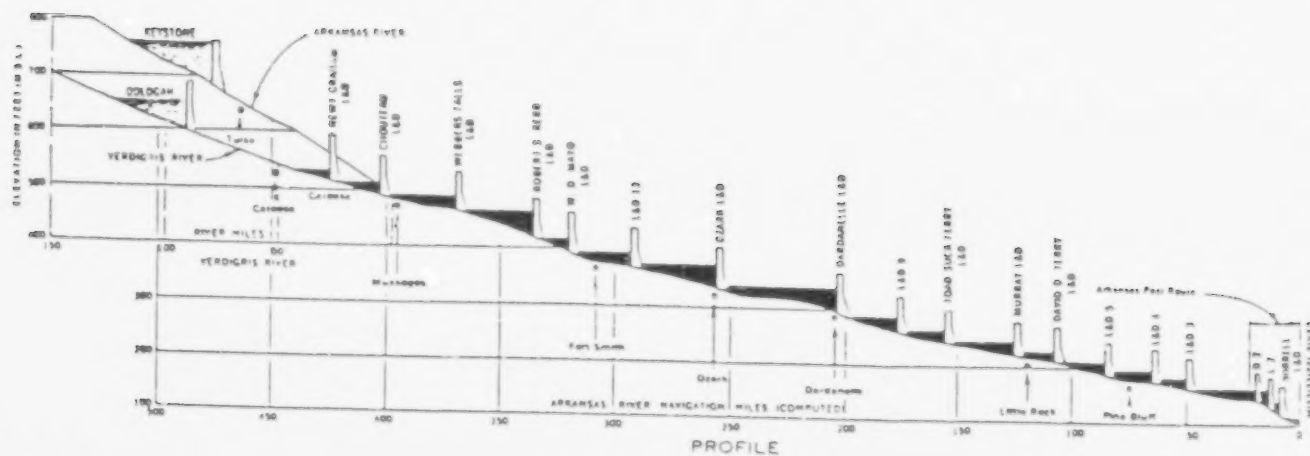


FIGURE 3-1 SCHEMATIC OF THE ARKANSAS RIVER NAVIGATION SYSTEM

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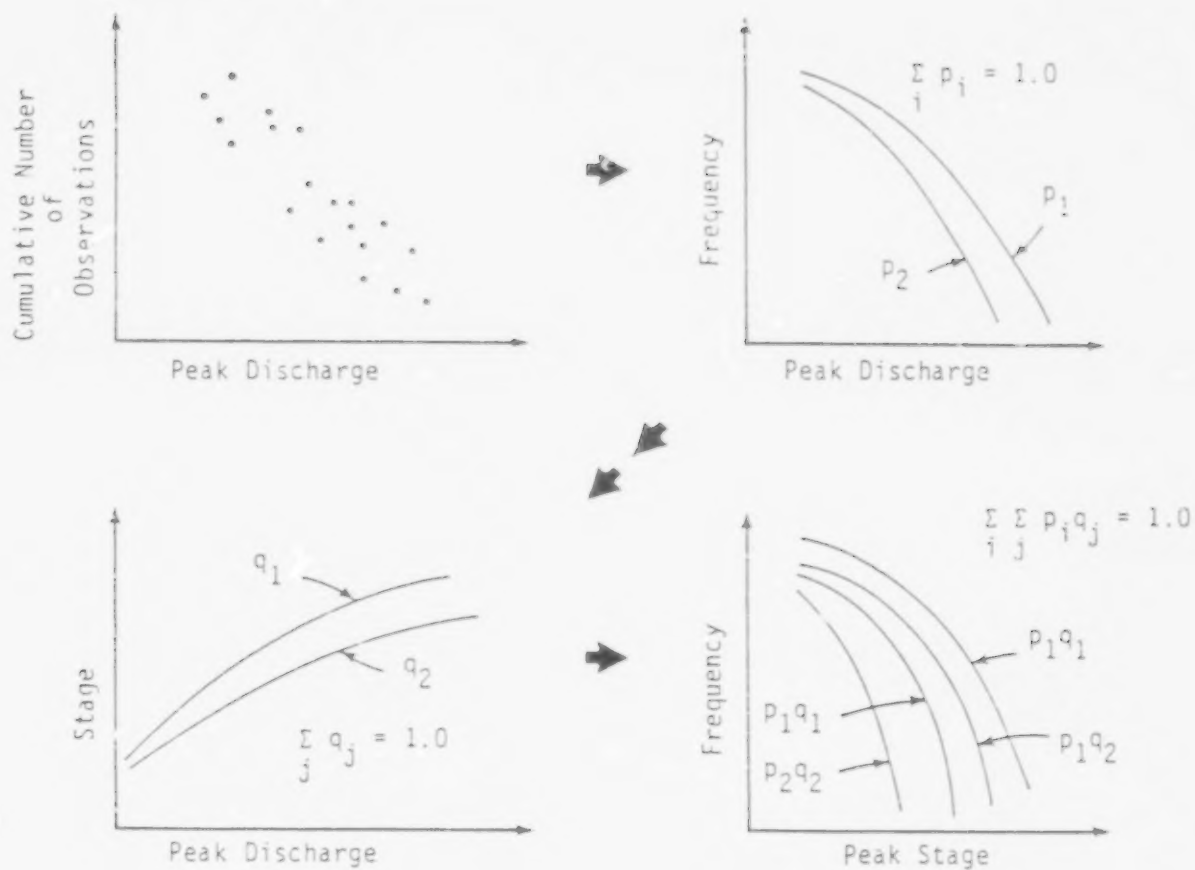


FIGURE 3-2 STEPS TO EVALUATE THE PROBABILITY DISTRIBUTION ON THE FREQUENCY OF FLOOD STAGE

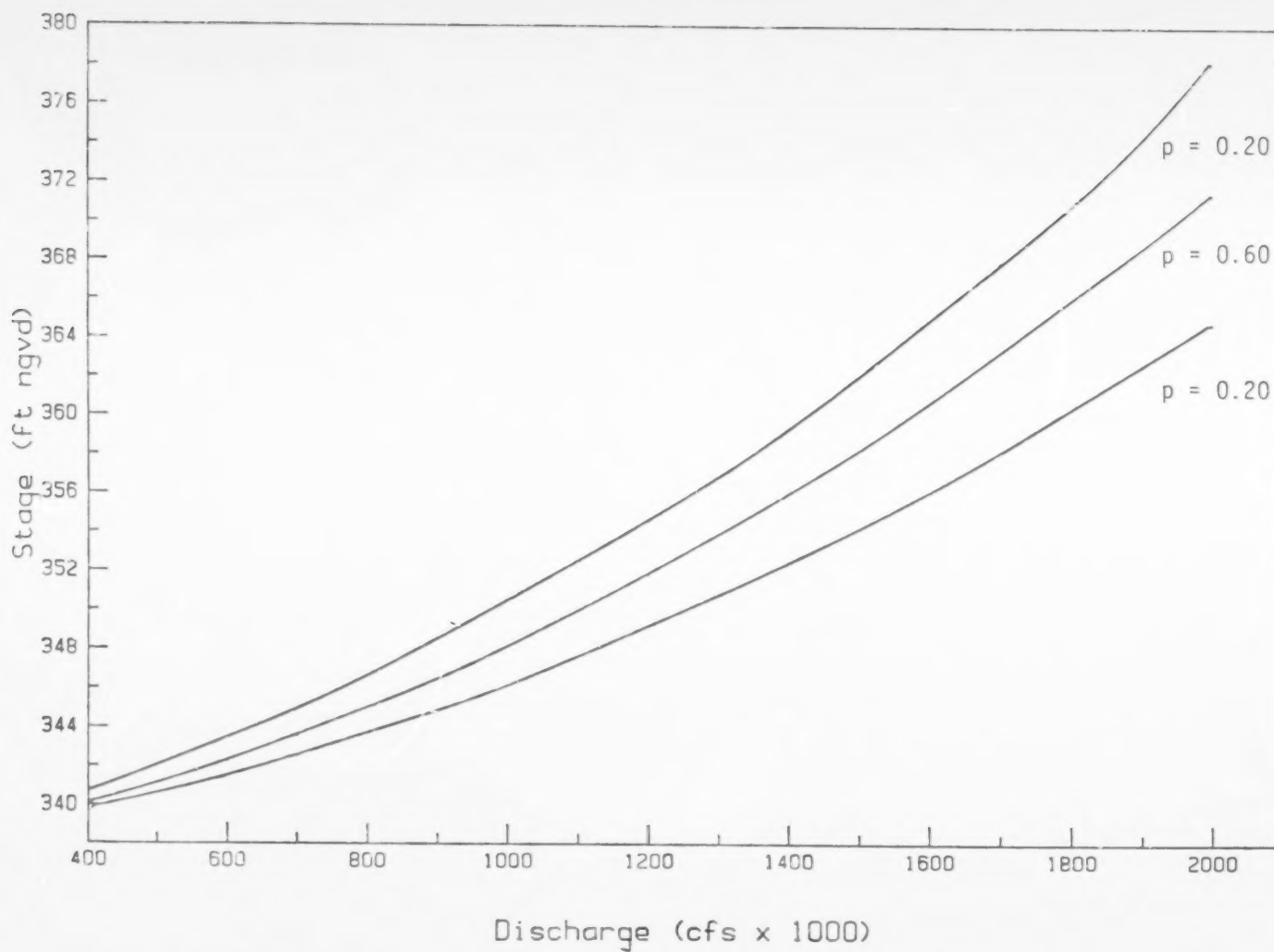


FIGURE 3-3 STAGE-DISCHARGE RELATION

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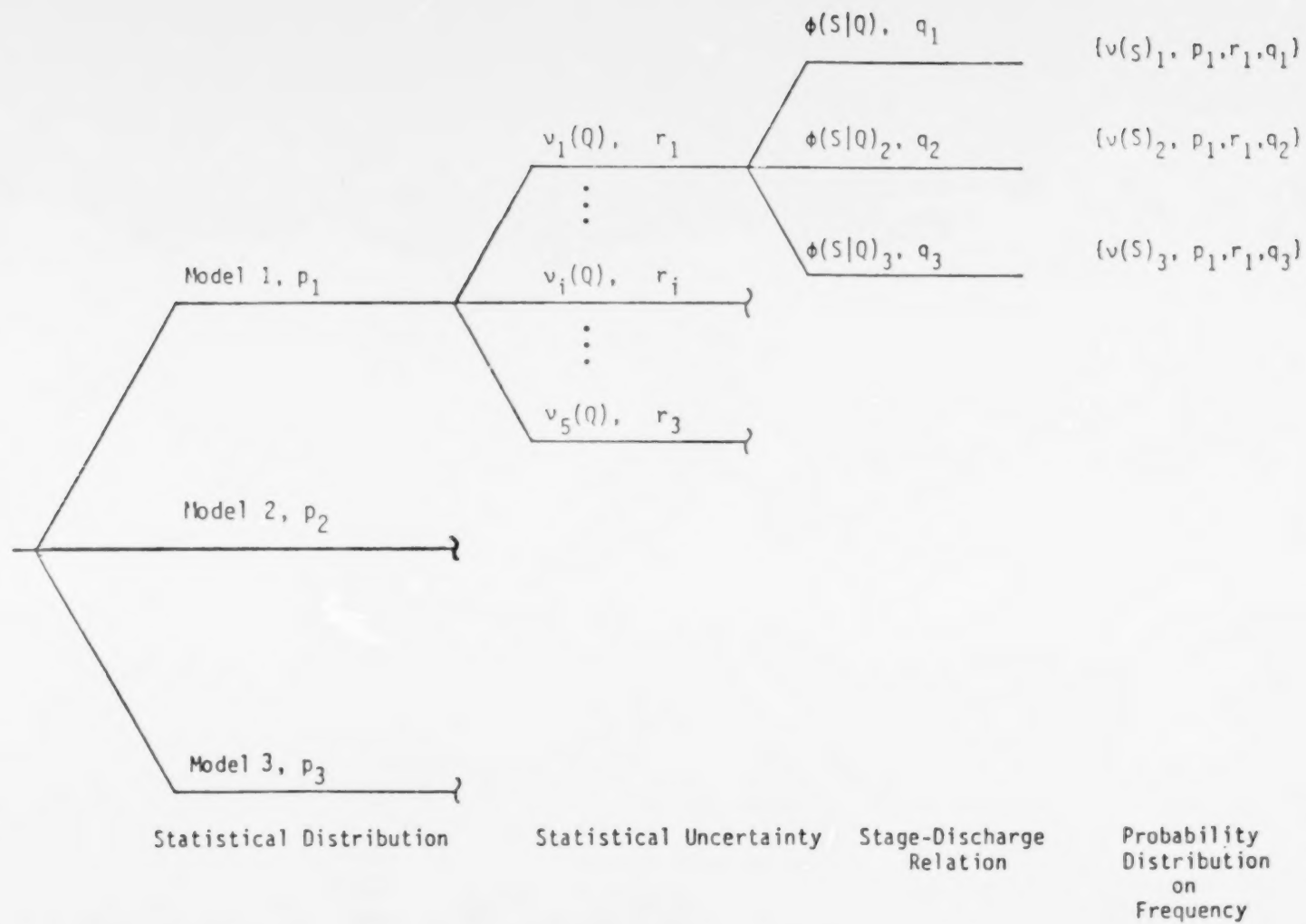


FIGURE 3-4 LOGIC DIAGRAM ILLUSTRATING THE FLOOD HAZARD UNCERTAINTY ANALYSIS

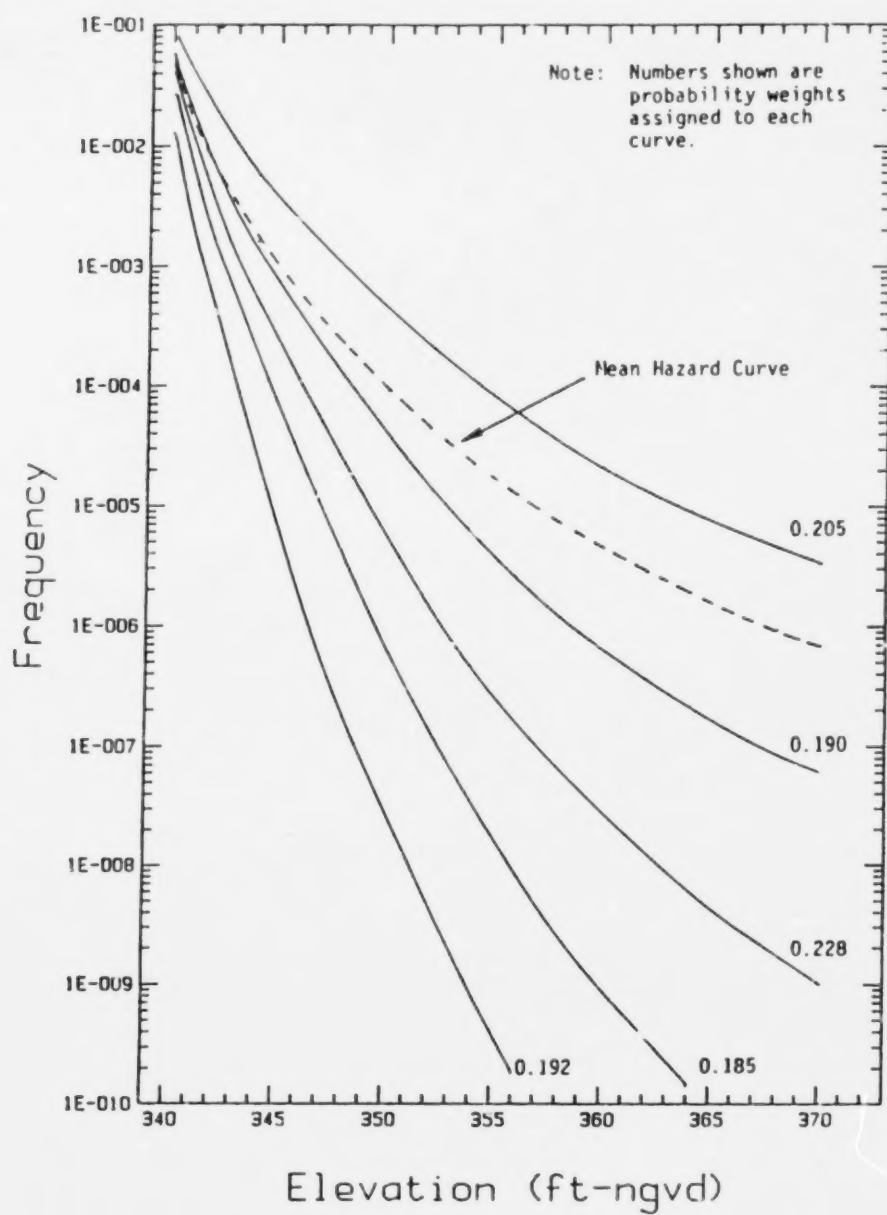


FIGURE 3-5 FLOOD HAZARD CURVES

4.0 FRAGILITY ANALYSIS

In this section fragility data for structures and equipment items vulnerable to the effects of flooding are developed. As indicated in Section 1.0, a conservative approach was used in developing capacities of structures and equipment to resist the effects of flooding. If it is found in the systems analysis that external flooding is a significant contributor relative to other external and internal initiators, a more detailed look at components which dominate the flood contribution may be required.

In Section 4.1 the methodology to evaluate the fragility of structures and equipment items is described. For the components listed in Table 2-2 which are vulnerable to the effects of external floods, fragility data are reported in Section 4.2.

4.1 Methodology

In this section the approach used to evaluate fragility data for structures and equipment items is described. The fragility assessment for plant structures is described first.

Structures

To estimate the fraction of failure of plant structures due to flood loads, an approach similar to that used in seismic analysis was adapted in which the capacity of a structure is expressed as a lognormal distribution. The parameters of the model that define the fraction of failure due to flood forces are:

- Median capacity defined in terms of flood elevation or depth

- Logarithmic standard deviation for capacity due to randomness, β_r
- Logarithmic standard deviation of the median capacity due to uncertainty, β_u

The median is a best estimate of the capacity of a structure. The fraction failure at this level is 0.50.

The two logarithmic standard deviations, β_r and β_u characterize the various sources of variability that exist in predicting structural failure. The first term, β_r , corresponds to the inherent randomness in the capacity of a structure to resist loads. An example is the natural variability in material properties. The second term, β_u , expresses the uncertainty in estimating the median capacity of a structure, or the location about which the fragility curve is centered.

The fragility of a structure is defined by the lognormal distribution which is expressed as:

$$F(d) = \Phi \left[\frac{\ln(d/D)}{\beta_r} \right] \quad (4.1)$$

where: $\Phi[]$ = the standard cumulative normal distribution
 $F(d)$ = fraction of failure evaluated at flood depth, d
 D = median capacity of the structure

In the above expression, D , is also uncertain, as defined by β_u . Thus, a family of fragility curves can be defined for a set of possible estimates of D .

A number of factors contribute to the capacity of a concrete structure for the effects of flood loads. They include:

- hydrostatic and dynamic wave forces
- pattern of cracking and failure
- failure level of a structure.

Fragility parameters are based on design information when possible or simplified calculations.

To access the capacity of plant structures, the fragility evaluation is made in terms of the depth of flooding at the plant site.

Equipment Items

Failure of components is assumed to occur due to submergence. Therefore, for each component a critical flood depth must be reached in a room in order to render it unavailable. In the case of inundation of the plant site due to river flooding, the fragility of equipment must ultimately be expressed in terms of global site inundation. This differs from an internal flood assessment where critical depths are expressed in terms of the depth of water in a room.

For equipment items located in rooms below plant grade, submergence can occur in two ways. First, for flood levels at or below grade, if there is cracking of a building's exterior walls, leakage will occur. The likelihood of this occurring is a function of the structural capacity of the walls. Flooding of the lower elevations can also occur if the site is inundated to levels higher than plant flood protection, in which case flood waters will propagate throughout the plant, flooding it to an elevation equal to the river stage.

To evaluate the fragility of equipment items exposed to external flooding, the following steps were taken:

- Estimate the depth of submergence necessary to fail a component in a room
- Identify flood protection devices (if any)
- Determine modes of flooding a room or area (i.e., propagation through doorways, down stairs, etc.)
- Transform local fragility values or critical depth estimates to a global flood hazard characterization

For each component vulnerable to flooding these steps were taken and fragility data estimated.

4.2 Results

Structures

Given the occurrence of an extreme flood, plant structures may be subjected to hydrostatic loads. According to the FSAR (Ref. 2), a water table at elevation 338 feet is assumed for normal operating conditions. However, the Auxiliary Building was checked against failure for the maximum flood elevation of 361 feet NGVD.

The effect of extreme hydrostatic loads on the exterior walls was evaluated by means of a finite element analysis (Ref. 18). According to these calculations, a flood elevation of 361 feet NGVD nearly equals the capacity of the exterior walls to resist hydrostatic forces. If the capacity of the walls is exceeded, it is assumed that the critical failure mode is cracking, which leads to in leakage into rooms at the lower elevations in the plant. Based on the information in Reference 18 the fragility of the exterior walls of the Auxiliary Building was estimated. This result is reported in Table 4-1.

Equipment

Table 4-2 summarizes the fragility data for safety equipment. In estimating the fragility of equipment located in non-watertight rooms below plant grade, it was assumed that the plant site must be inundated to an elevation of 361 feet NGVD. This is 8 feet above plant grade and 7 feet above the elevation of the ground floor in plant structures.

In Table 4-2 the mode of flooding for each equipment item is identified. Two designations are used, site inundation and building leakage. The first denotes a mode in which the plant site must be inundated in order to flood the location of the particular equipment item. The second refers to cracking of the exterior walls of the Auxiliary Building due to hydrostatic load.

TABLE 4-1
STRUCTURE FRAGILITY ESTIMATES

Structure	Median* (feet)	β_r	β_u	β_c
Auxiliary Building	14.0	0.10	0.25	0.27

*Fragility curve is calculated in terms of the depth of water above elevation 348 feet NGVD.

TABLE 4-2
FRAGILITY DATA FOR EQUIPMENT ITEMS

Equipment	Mode of Flooding	Fragility Data (feet NGVD)	Building Leakage
Emergency Switchgear	Site inundation	372.0	NA
480V Switchgear	Site inundation	372.0	NA
Emergency Feedwater Pumps	Building leakage/ Site inundation	361.0	See Table 4-1
HPIS Pumps	Building leakage/ Site inundation	361.0	See Table 4-1
LPIS Pumps	Building leakage/ Site inundation	361.0	See Table 4-1
Diesel Generators	Site inundation	369.0	NA
Battery Room	Site inundation	372.0	NA
Cable Spreading Room	Site inundation	374.0	NA
Control Room	Site inundation	386.0	NA
Computer Room	Site inundation	386.0	NA
RBSS Pumps	Building leakage/ Site inundation	361.0	See Table 4-1
Service Water Pumps	Site inundation	367.5	NA
Switchyard	Site inundation	354.0	NA

5.0 SYSTEM ANALYSIS

External flooding can potentially cause failure of the safety systems in the plant. In this section the response of the plant to the safety system failures is discussed. A procedure is given that is used in Section 6.0 for quantifying risk.

The methodology used to perform the systems analysis is given in Section 5.1. The results are given in Section 5.2.

5.1 Methodology

The principal cause of flooding at ANO-1 that could potentially cause failure of the plant's safety systems is river flooding due to runoff with coincident failure of the Ozark dam.

To determine the contribution this river flooding can have to core melt probability, the following steps are performed:

- (1) The flood level that will fail sufficient safety systems resulting in core melt is determined.
- (2) The frequency of an external flooding event that will result in the above flood level is determined.
- (3) The first two steps will give the probability of core melt due to external flooding if no actions (recovery or otherwise) are taken by plant personnel. This gives an upper bound result. The fact that procedures have been established to shutdown the plant in the event of severe flooding are discussed in this analysis.

5.2

The results for external flooding systems analyses are given in the following two subsections.

5.2.1 Component/System Failures Due to External Flooding

The systems that will fail as a result of the failure of the various equipment/structures are shown in Table 5-1. The critical flood elevation is also given for the various components. The vulnerable equipment/structures listed in the left column are obtained from Table 2-3. The critical flood level elevations are obtained from Table 4-2. The fact that a LOSP will occur is included in determining which systems will fail. This is because the flood level of the switchyard is lower than the rest of the systems. Each system failure is discussed below.

(1) All Safety Systems due to flooding of Emergency Switchgear

Flooding of the emergency switchgear is assumed to fail all the power to safety systems.

(2) All Safety Systems due to flooding of 480V Switchgear

Flooding of the 480V Switchgear would fail all the 480VAC power to the safety systems. This would fail all the ECI systems. Only the EFWS turbine train could operate, however, it would eventually fail due to battery depletion. This is because the failure of the 480VAC buses would lead to loss of power to the DC buses that ^{would} ~~provide~~ instrumentation and control power to the turbine driven pump.

(3) EFWS due to flooding of the EFWS pumps

The flooding of the EFWS pumps will fail the EFWS. This will cause loss of secondary cooling for the plant. The PCS will fail due to the simultaneous LOSP.

(4) HPIS due to flooding of the HPIS pumps

The flooding of the high pressure injection system (HPIS) pumps will fail the HPIS which would be needed for feed and bleed.

TABLE 5-1
SYSTEMS FAILED AS A RESULT OF EQUIPMENT/STRUCTURE
EXTERNAL FLOODING AT ANO-1

Vulnerable Equipment/Structure Failed	Critical Elevation (Feet-NGVD)	Systems(s) Failed ¹
1. Emergency Switchgear	372.0	All Safety Systems
2. 480V Switchgear	372.0	All Safety Systems
3. Emergency Feedwater Pumps	361.0	EFWS
4. HPIS pumps	361.0	HPIS
5. LPIS pumps	361.0	LPIS
6. Diesel Generators	369.0	HPIS/LPIS/HPRS/LPRS/EFWS ²
7. Battery Room	372.0	All Safety Systems
8. Cable Spreading Room	374.0	NONE
9. Control Room	386.0	All Safety Systems
10. Computer Room	386.0	NONE
11. RBSS pumps	361.0	RBSI/RBSR
12. Service Water Pumps	367.5	SWS & all safety systems except EFWS and the diesel generators
13. Switchyard	354	LOSP

- ¹ HPIS = High Pressure Injection System
 LPIS = Low Pressure Injection System
 HPRS = High Pressure Recirculation System
 LPRS = Low Pressure Recirculation System
 EFWS = Emergency Feedwater System
 RBSS = Reactor Building Spray System
 RBSI/R = Reactor Building Spray Injection/Recirculation System
 SWS = Service Water System

- ² The EFWS will also fail. The motor driven pump will fail due to LOSP. The turbine driven pump could fail due to battery depletion, random failures, test or maintenance, CST depletion, etc.

(5) LPIS due to flooding of the LPIS pumps

The flooding of the low pressure injection system (LPIS) pumps will fail the LPIS.

(6) HPIS/LPIS/HPRS/LPRS/EFWS due to flooding of the diesel generators

Flooding of the diesel generators will only fail these systems if there is a simultaneous loss of offsite power (LOSP) during the external flooding event. Both early (HPIS/LPIS) and later (HPRS/LPRS) emergency coolant injection systems will fail. The EFWS turbine driven train will also fail due to battery depletion or can fail randomly.

(7) All Safety Systems due to flooding of the battery room

None of the safety systems can function without DC power via the batteries in the event of transient with LOSP.

(8) Flooding of the cable spreading room

It is assumed that the cable spreading room is capable of withstanding a flood environment

(9) All Safety Systems due to flooding of the control room

The control room is essential to shutting down the plant. However, in this case it is a moot point since the level of the control room (386' NGVD) is 24 feet above the level that will result in flooding of all the safety systems that could prevent core melt.

(10) Flooding of the computer room

Due to the elevation of the computer room, its flooding is a moot point. However, it is not considered to be important in the prevention of core melt.

(11) RBSI/RBSR due to flooding of the RBSS pumps

The reactor building spray injection system (RBSI) and its recirculation mode (RBSR) are not required to prevent core melt.

(12) SWS due to flooding of the service water pumps

The flooding of the service water system (SWS) pumps is assumed to fail the SWS. This would fail all the safety systems except the EFWS turbine train and the diesel generators.

(13) LOSP due to flooding of the switchyard

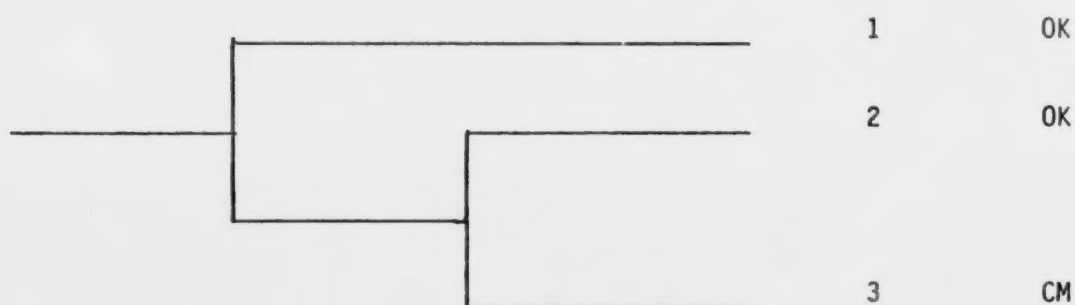
Flooding of the switchyard will result in LOSP.

As noted earlier, a flood level of 361 feet NGVD will fail all the safety systems including the EFWS. At this flood level, the assumption of a simultaneous loss of offsite power is not required to fail any system.

5.2.2 External Flooding Event Tree

A functional event tree that models the response of ANO-1 to external floods is shown in Figure 5-1. This tree assumes that a simultaneous LOSP occurs at the time of the external flooding. In Boolean equation form, the functions are systemically defined as the following:

Function	Secondary Cooling (SEC)	Emergency Coolant Injection (FNB Mode)	Sequence Number	Core Status
System(1) Support Systems	EFWS	HPIS		



- (1) EFWS = Emergency Feedwater System
 HPIS = High Pressure Injection System

FIGURE 5-1. EXTERNAL FLOODING FUNCTIONAL EVENT TREE FOR ANO-1

Secondary Cooling = EFWS

where EFWS = Emergency Feedwater System

$$ECI = HPIS * LPIS + HPRS * LPRS$$

where: ECI = Emergency Coolant Injection

HPIS = High Pressure Injection System (Feed and Bleed (FNB) Mode)

*LPIS = Low Pressure Injection System

*HPRS = High Pressure Recirculation System

*LPRS = Low Pressure Recirculation System

Under each function in the event tree in Figure 5-1 the system(s) whose flooding will cause the function to fail is listed.

*These systems would only be needed for long term cooling. The only part of ECI that is included in this analysis is the feed and bleed mode using HPIS.

6.0 RISK QUANTIFICATION

6.1 Methodology and Results

In this section the core melt frequency contribution at ANO-1 due to external flooding is determined. This is done by using the system failure information and functional event tree developed in Section 5.0.

As can be seen from the functional event tree given in Figure 5-1, and from the analysis of the systems that will fail due to an external flood in Table 5-1 only one flood level is important: 361-feet NGVD.

At 361 feet NGVD all the safety systems will fail. Core melt is expected to commence in approximately 30 minutes. Recovery of offsite power within 30 minutes is not considered since all the pumps (EFWS, HPIS and LPIS) are submerged; therefore, the possibility of any recovery actions is highly unlikely. The frequency of a 361 feet flood is $7.2E-6/\text{year}$.

No credit is taken for operator actions, even though, emergency procedures have been established to shut the plant down should the Arkansas River exceed elevation 350 feet. This is because it is not known how long it would take to reach 361 feet NGVD from 350 feet. In addition, even though the plant was shutdown, there still would be some residual heat that would need to be removed and no safety systems would be available to remove the heat.

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APPENDIX G

EXTREME WIND ANALYSIS FOR THE
ANO-1 NUCLEAR POWER PLANT

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1.0 INTRODUCTION

1.1 Background

Sandia National Laboratories (SNL) is participating in the Task Action Plan (TAP) A-45 program to evaluate the adequacy of current licensing requirements for shutdown decay heat removal systems. As part of the SNL study, the response of decay heat removal systems to internal and external initiating events is being considered. The full scope of the analysis plan is reported in Reference 1.

Science Applications International Corporation (SAIC) is assisting SNL in the internal event and external event analyses being conducted for the U.S. Nuclear Regulatory Commission (USNRC). Jack R. Benjamin and Associates, Inc. (JBA) is a subcontractor to SAIC and provided the wind hazard and fragility data used in the TAP A-45 risk analyses. The analyses involving extreme winds for the Arkansas Nuclear One, Unit 1 (referred to as ANO-1) are given in this report.

1.2 Extreme Wind Risk Analysis

As stated in Reference 1, it is beyond the scope of the TAP-45 risk program to perform an in-depth probabilistic risk assessment. Instead, the objective is to conduct an analysis that quantifies the significant threats to a plant. However, the analysis procedure used to perform the wind analysis embodies the basic philosophy of a full-scope probabilistic risk assessment.

To evaluate the frequency of plant damage due to external wind events, the following tasks are performed: 1) plant familiarization, 2) tornado and straight wind hazard analyses, 3) tornado missile and wind pressure fragility analyses, 4) systems analysis, and 5) risk quantification. Figure 1-1 shows the relationship of the steps in the wind risk analysis. The hazard and fragility analyses are being conducted by JBA while the systems and risk quantification are being performed by SAIC.

The steps in the analysis are summarized below:

- o Plant Familiarization - The purpose of this task is to gather information related to the occurrence of extreme wind hazards at a plant and the vulnerability of plant structures and equipment to wind effects. In assessing the reliability of shutdown decay heat removal systems, a review of available documentation (i.e., FSAR, plant drawings and topical reports) and a walkdown of the plant are performed.
- o Hazard Analysis - Mean wind hazard curves for tornadoes and straight winds were developed by the USNRC for use in the TAP A-45 project. The purpose of the hazard analysis conducted by JBA is to develop the uncertainty in the hazard curves. Independent tornado hazard analyses are conducted which include consideration of the uncertainty in the tornado plant strike model, site size, tornado data, tornado damage area and length variation, and F-scale intensity/windspeed conversion. The resulting curves are scaled so that the resulting mean frequencies of exceedance equal the USNRC values. An analysis of hurricanes and other straight winds was conducted to develop the uncertainty in a similar manner as was done for tornadoes. Uncertainty in the windspeed data bases, applicability of recording station data to the site and terrain roughness differences are considered.
- o Tornado Missile and Wind Pressure Fragility Analyses - A tornado missile analysis was conducted by JBA which considers the occurrence frequency of tornadoes, number of available tornado missiles, target exposure, and target area. Results of past tornado missile impact studies which were based on detailed missile simulations are used. For structures and equipment vulnerable to wind pressure hazards, the cumulative fraction of failure defined as a function of windspeed, called a fragility curve, was assessed. Due to uncertainty, multiple fragility curves are determined for each component.
- o Systems Analysis - The shutdown decay heat removal systems analysis was performed by developing event trees and fault trees with external wind as the initiating event. The component fragilities and missile impact factors are used to evaluate the conditional frequency of failure of different safety systems.

- o Risk Quantification - The quantification of the likelihood of plant damage is assessed by properly weighting the conditional fraction of plant damage (i.e., system fragility curve), defined as a function of wind intensity, by the frequency of occurrence of each windspeed. By propagating the uncertainty in the frequency of wind hazards and the component fragilities, the probability distribution on the frequency of core melt is evaluated.

1.3 Analysis Approach

A conservative approach was used in developing the capacities of the structures and equipment to resist the effects of wind. This procedure is based on the a prior assumption that the individual contribution of each of the components to core melt is not significant. If it is found in the systems analysis that wind hazard is a significant contributor relative to other external and internal initiators, then a more detailed look at components which dominate the wind contribution will be required. By proceeding in this manner the analysis effort is focused on the most significant contributors, and components which are not contributors are screened out by a defensibly conservative approach that requires a minimum of effort.

1.4 Report Organization

In Section 2.0, the plant and site characteristics of ANO-1 are described from the perspective of the effects of wind events on shutdown decay heat removal systems. The plant site is described and the critical structure and equipment items vulnerable to wind effects are given.

In Section 3.0, the uncertainty analysis for tornado and straight winds, is presented. Hazard data and corresponding curves are given.

Sections 4.0 and 5.0 present the results of the tornado missile and wind pressure fragility analyses, respectively. Fractions of missile impact and wind pressure fragility distribution parameter values are given.

In Section 6.0 the system analyses are presented. A methodology is developed that is used to analyze the plant's response to various system failures.

In Section 7.0 the risk quantification is presented. Core melt probability contributions are given for straight winds, tornado winds and tornado missiles.

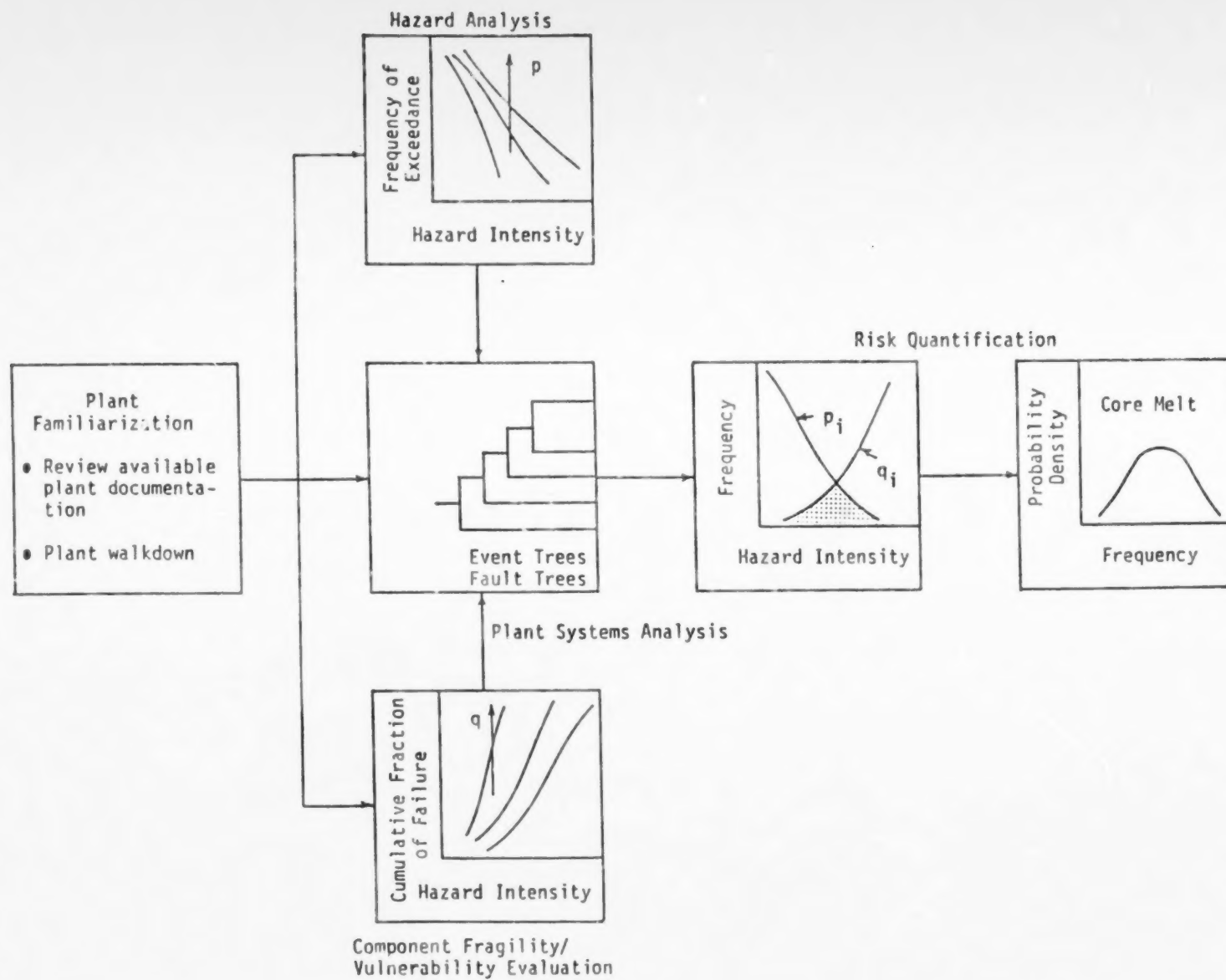


FIGURE 1-1 RISK ANALYSIS FLOWCHART

2.0 PLANT AND SITE CHARACTERISTICS

A plant walkdown of ANO-1 was conducted to gain familiarity with the plant and site characteristics and to determine the vulnerable structures and equipment. Copies of criteria and design documents, and construction drawings were obtained and reviewed. Discussions with plant and utility personnel were conducted to obtain an understanding of the structural protection provided to resist the effects of wind.

A description of the plant and site characteristics in regard to the wind analysis is given in Section 2.1. A list of vulnerable structures and equipment which were analyzed are discussed in the following section. Finally, a general description of the wind hazards at the ANO-1 plant site is described in the last section.

2.1 Plant Description and Design Wind Loadings

The structures at ANO-1 are constructed of reinforced concrete or steel framing with metal siding. Figure 2-1 is a plot plan of the site which shows the general arrangement of the structures and location of equipment vulnerable to wind effects. These include the structures and equipment listed in Table 2-1.

The safety-related equipment which is part of the shutdown decay heat removal system is contained within the containment, auxiliary building, turbine building, and intake structure. The condensate storage tanks (CST), the borated water storage tank (BWST) and the diesel fuel oil storage tank are located at ground level outside of the buildings (Figure 2-1). The two diesel generator stacks pass through the auxiliary building and are exposed above the roof of the building.

In general, the safety-related equipment is protected in Class 1 structures with exterior concrete walls and slabs at least 12 inches thick. In some locations the walls and roof slabs are thicker. One exception is the diesel generator exhaust stacks which are anchored to the Class 2 metal siding of the auxiliary building.

All structures at ANO-1 were designed as a minimum for a loading corresponding to a 67 mph windspeed (Ref. 2). In addition, the Class 1 structures, which include all the structures that contain the decay heat removal equipment, are designed for tornado loads stated in Reference 2 as follows:

- "1. A tangential velocity of 300 mph.
2. A forward progress of 60 mph.
3. A pressure drop of three psi occurring over a three-second time interval."

In addition, the structures were designed for the following tornado missile effects (note that the tornado pressures were not combined with the missile effects):

- (1) A 4-inch x 12-inch wooden plank, 12 feet long, weighing 160 pounds traveling end-on at 300 mph.
- (2) A 4000-pound automobile flying through the air at 50 mph and at not more than 25 feet above the ground with a contact area of 20 square feet.
- (3) A 3-inch diameter Schedule 40 pipe, 10 feet long traveling end-on at 100 mph.

No evidence was found to indicate that a wind pressure or tornado missile analysis was conducted for any of the exposed equipment (e.g., see Table 2-1).

2.2 Vulnerable Structures and Equipment

In general, most safety-related equipment is contained within and protected by concrete structures which are at least 12 inches thick, and in many cases the barriers appear to be thicker. Equipment which is not protected and which is vulnerable to wind effects is listed in Table 2-1. It was assumed in the development of the vulnerable structure and equipment list that concrete walls and roof slabs greater than 12 inches thick provide adequate protection against tornado missiles. Doors, blockouts, exhaust stacks and other openings to the Category I structures were postulated to be potential vulnerable entrances to the buildings for tornado missiles. In general, all openings were either found to be protected by barrier walls, tornado missile barriers or too small to be significant contributors to the frequency of core melt. Based on a scoping analysis, openings less than 100 ft² correspond conservatively to a mean frequency of tornado missile impact less than 5.0-7¹ per year; thus, the door openings were not found to be significant contributors and were not considered further.

The BWST, the CSTs and the diesel oil storage tank are located outside and are vulnerable to wind effects. The tanks were not analyzed in this study for wind pressure effects based on experience with similar tanks at other nuclear power plants. Because the newly built 321,000 gallon tank was designed for seismic loads, the capacity of this tank is sufficient to resist wind velocities greater than 360 mph. The other CST is not seismically designed but probably has a high wind pressure capacity. However, these tanks are both vulnerable to tornado missiles and were included in the tornado missile analysis.

The exhaust stacks which go from the diesel generators up through the auxiliary building to the roof are exposed to tornado missiles above the roof. As indicated in the FSAR, the auxiliary building is designed

¹ 5.0-7 = 5.0 x 10⁻⁷

to resist winds up to 300 mph; however, the part of the auxiliary building where the exhaust stacks are attached consists of metal siding which is assumed to be Class 2 construction.

The safety-related buildings and structures at ANO-1 have been designed to resist wind pressures at least equal to 300 mph. This windspeed corresponds to a mean frequency of occurrence of approximately 2.0-6 per year. Because of conservatism in the design, the capacity is likely to be significantly higher and the corresponding mean frequency even lower. Based on this result the capacities of the buildings were not considered further. Thus, no detailed wind pressure analysis was required for the Class 1 part of the ANO-1 Nuclear Plant.

2.3 Wind Hazard

The ANO-1 site is exposed to potential winds generated by tornadoes and extratropical cyclones (including winter storms and summer thunderstorms), which are referred to in this report as straight winds. Because the site is located in the central portion of the U.S., there is no threat from hurricane winds.

A listing of all tornadoes within approximately a 145-mile radius of ANO-1 was obtained from the National Severe Storms Forecast Center (NSSFC) in Kansas City, Missouri. From 1950 through 1983, a total of 965 tornadoes were recorded of which 23 were rated Fujita scale F-4 (i.e., 207 to 260 mph) and one was rated F-5 (261 to 318 mph). Figure 2-2 shows the traces of F-4 and F-5 tornadoes in the area near the site based on a record between 1880 and 1982 (Ref. 3). The location of the F-5 event which is within a radius of 145 miles of the site as listed in the NSSFC database is indicated in Figure 2-2.

ANO-1 is located in Region I of the USNRC tornado risk regionalization scheme given in WASH-1300 (Ref. 4). This is the region which has the highest hazard of the three USNRC regions. It is also

located in Region A of the 4-region scheme proposed by Twisdale and Dunn as shown in Figure 2-3 (Ref. 5). Region A has a regional occurrence rate corrected for unreported tornadoes (occurrences per square mile per year) of 5.179-4 compared to USNRC Region I which has an occurrence rate of 4.12-4. The percentage of large tornadoes (i.e., F-4 and F-5s) is also lower in USNRC Region I compared to Region A. Although Region B has a slightly greater total tornado occurrence rate per square mile per year (6.98-4) than Region A, Region A has a greater percentage of large events (i.e., $\geq$ F-4).

Straight wind data were obtained from the weather station at Little Rock, Arkansas which is approximately 57 miles southeast from the site (Ref. 6). For the period 1952 to 1977 the largest recorded fastest mile windspeed is reported to be 61 mph, which occurred in 1974.

TABLE 2-1
VULNERABLE STRUCTURES AND EQUIPMENT

Structures/Equipment	Location	Vulnerability
Condensate Storage Tank (CST)	Southwest of Containment Building	Vulnerable to tornado missiles
Borated Water Storage Tank (BWST)	West of Auxiliary Building	Vulnerable to tornado missiles
Diesel Fuel Oil Storage Tank	Northeast of Cooling Tower	Vulnerable to tornado missiles
New 321,000 Gal. Storage Tank (CST)	West of Auxiliary Building	Vulnerable to tornado missiles (50,000 gallons of water are protected from missile impact by 5' high concrete wall).
Diesel Generator Exhaust Stacks	Auxiliary Building	Vulnerable to tornado missiles and wind pressure effects

* See Figure 2-1 for general arrangement of buildings

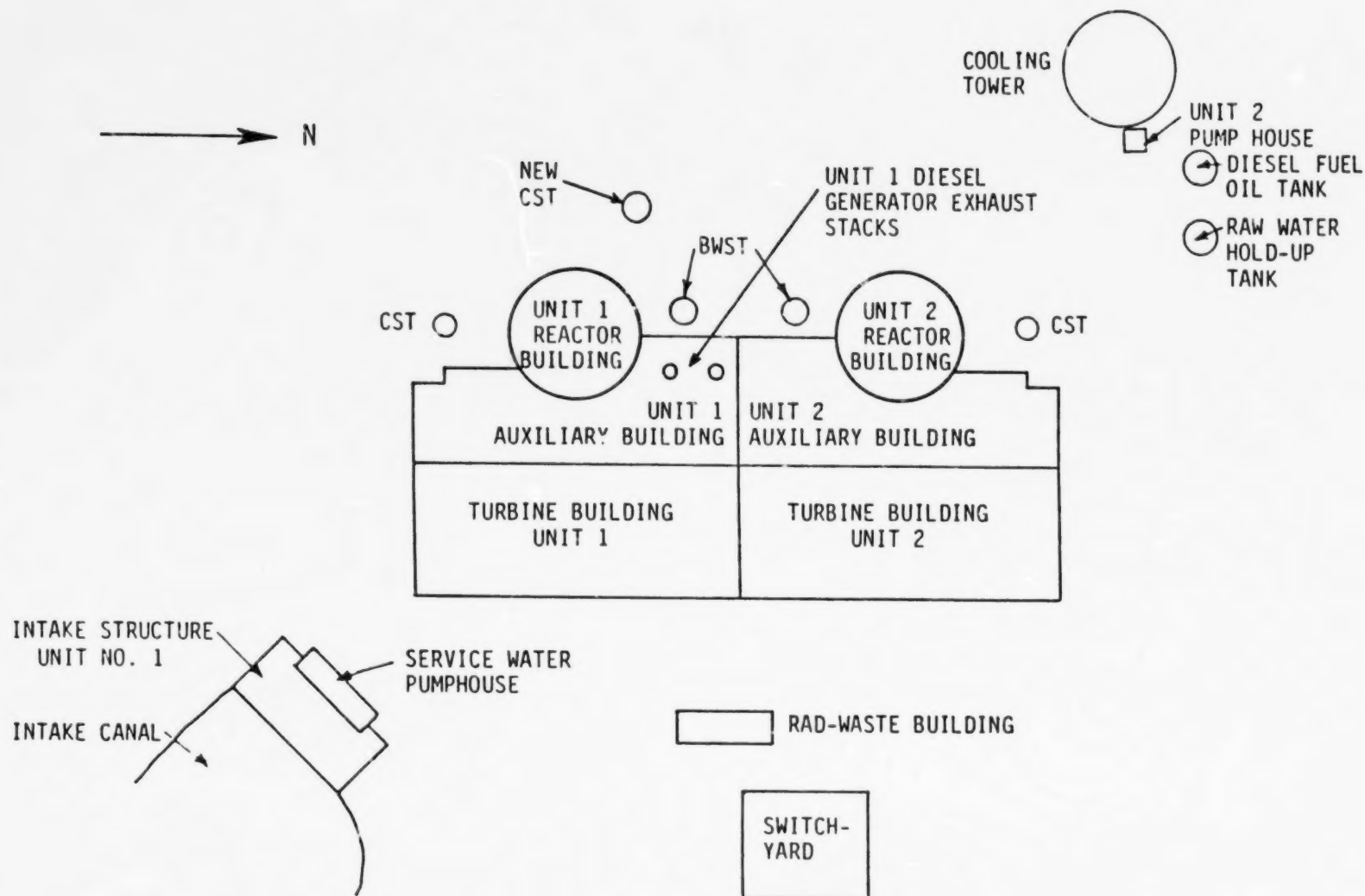


FIGURE 2-1 ANO-1 PLOT PLAN

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FIGURE 2-2 HISTORIC F-4 AND F-5 TORNADO TRACES (Ref. 3)

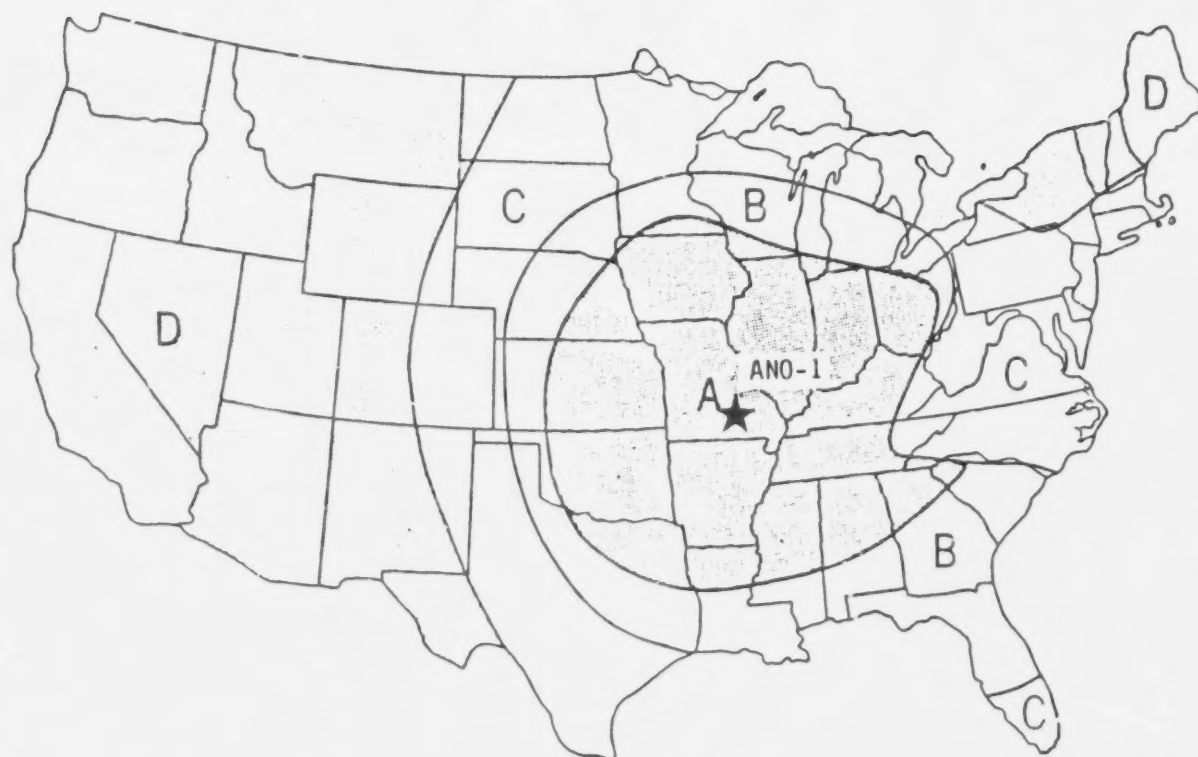


FIGURE 2-3 TORNADO RISK REGIONALIZATION SCHEME PROPOSED BY TWISDALE AND DUNN (Ref. 5)

3.0 WIND HAZARD

Hazard data were developed for both tornadoes and straight winds for the ANO-1 site. Based on an agreement between the USNRC and SNL, the Meteorology Section, Meteorology and Effluent Treatment Branch of the USNRC, prepared best estimate hazard curves for the ANO-1 site for both wind types. The basis for this agreement was that the Meteorology Section had an established procedure for determining the frequency of high winds at nuclear power plant sites and that the analyses to be provided by SNL should utilize those frequency assessments to be consistent with other risk analyses requiring wind hazard evaluations.

The objective of the wind hazard analysis performed for ANO-1 by JBA was to develop the uncertainty about the best estimate hazard curves which were assumed to be mean hazard curves (i.e., for tornado and straight winds) and to prepare a family of curves, which have a corresponding mean curve that is equal to the USNRC curve for each of the two wind types.

The next section discusses the methodologies used to calculate the uncertainty in the wind hazard. The last section presents the results of the hazard analysis.

3.1 Methodology

The methodology for the uncertainty analysis performed for tornadoes and straight winds is given in Sections 3.1.1 and 3.1.2, respectively.

3.1.1 Tornadoes

The tornado analysis followed the basic tornado analysis procedure given by Reinhold and Ellingwood (Ref. 9). Uncertainty was considered in the following five parts of the analysis:

- Strike model
- Plant site width
- Tornado analysis data
- Tornado damage area and length variation
- F-scale windspeed conversion

Multiple analyses were performed to develop a family of wind hazard curves. The mean frequency of exceedance was determined at seven windspeeds (i.e., 66, 93, 134, 182, 234, 290, and 349 mph) and compared to the corresponding mean frequency values developed by the USNRC. All hazard curves were shifted by the ratios of calculated mean values to the USNRC values. From the scaled probability distributions at each windspeed, five equally weighted frequency of exceedance values were calculated. The mean values of the resulting five frequencies at each windspeed are equal to the USNRC values and have a reasonable distribution which reflects the variabilities developed from a systematic consideration of the underlying uncertainties in the models and parameters.

Figure 3-1 shows the uncertainty analysis flowchart. The probability weights shown on each branch were subjectively assigned. The basis for the five parts of the analysis is discussed below.

Tornado Plant Strike Model

Two strike models were considered. Figure 3-2 shows the geometric relationships for the point strike and union strike models. The point strike model considers the site to be represented by a point and

requires that any part of the tornado which crosses the site leads to a strike. Thus the tornado strike area is equal to the area enclosed by the dashed centerline shown in Figure 3-2 which is simply equal to the tornado width, W , times the tornado length, L .

In contrast, the union strike model considers the site to be represented by a rectangular area of width, A , and length, B . Again if any part of the tornado crosses any part of the site, a strike occurs. The tornado strike area is equal to the area enclosed by the dashed centerline as shown in Figure 3-2, which is approximately equal to WL plus AL . Note that the areas equal to BW plus AB have been neglected since they are small relative to the area retained. By formulating the model in this manner the union strike model is equal to the sum of the point strike model and a "lifeline" term which is proportional to the width of the site.

Plant Site Width

The second part of the analysis shown in Figure 3-1 which was considered to be uncertain is the plant site width (see dimension A in Figure 3-2). Since the direction of a tornado strike is not known the width varies unless the site is a perfect circle. Three site widths were considered (200, 400 and 800 feet). The lower value simulates the possibility that the tornado must overlap the site for damage to occur, while the upper value conservatively includes all possible structures at the site. The middle value represents a diagonal across the area which contains the major safety-related structures.

Tornado Analysis Data

The tornado analysis data considered both local and regional data. As indicated in Table 3-1 the tornado analysis data consist of the occurrence fraction of F-scale tornado intensities, the average tornado area and average tornado length corresponding to each intensity, and

finally the rate of tornado occurrence in the area surrounding the site. The regional data corresponds to Region A, which contains the ANO-1 site, as shown in Figure 2-3. These data are given in Reference 9.

The local data were based on the tornadoes recorded in the NSSFC data base from 1950 through 1983 within a radius of 145 miles of the site (see discussion in Section 2). The tornado rate of occurrence and distribution of F-scale intensities were obtained from these data, but considered reporting trends, direct classification and random encounter errors, and unrated tornadoes (see Reference 9 for a discussion of these issues). The average tornado damage area and path length for the local area are based on the NSSFC data for intensities less than F-4. Since few F-5 events have occurred near the site, the regional data were used. Note that no F-6 tornadoes have ever been recorded, hence the F-6 data are based on extrapolation.

Tornado Damage Area and Length Variation

A tornado rated by the Fujita F-scale is based on the highest level of damage which occurs within the total tornado damage area. Figure 3-3 shows a sketch of a hypothetical F-4 tornado illustrating the variation of intensity within the total damage area. Note that only a small portion of an F-4 rated tornado is actually F-4 damage. There are various relationships which divide the total damage area into the individual lower F-scale damage areas. Table 3-2 shows three possible relationships. For each matrix shown the rows correspond to the actual damage state while the columns refer to the reported damage state. For example in the first matrix (Reinhold 1) of Table 3-2, 62.5 percent of the area of an F-6 rated tornado is actually F-0 damage.

The Reinhold 1 matrix is based on a variation of intensity along the tornado length using the percentage of damaged length for each tornado analyzed along with an empirically derived damage width relationship. This matrix is considered to be the most realistic. The

Reinhold 2 matrix considers the variation of intensity along the tornado length based on dividing the tornadoes into 2 to 3 mile segments and assigning the highest intensity observed in each segment as the intensity of the whole segment. The damage width relationship for the second matrix is based on a theoretical wind model with a tornado translation speed of 59 mph.

The third relationship considers the entire damage to be equal to the maximum damage (very conservative). This approach was used in WASH-1300 (Ref. 4) and is assumed in the USNRC procedure for determining wind hazard.

Table 3-3 gives the corresponding matrices for the damage length relationships used for the lifeline term calculation. The corresponding assumptions as discussed for the area relationships were used in the development of the matrices given in Table 3-2.

F-Scale Windspeed Conversion

The last part which was varied in the uncertainty analysis is the conversion between the F-scale intensities and the maximum windspeed. There have been discussions in recent years that the windspeeds originally suggested by Fujita are too high (Refs. 8 and 9). In addition to the Fujita conversion, two additional conversions were assumed as given in Table 3-4. These were developed by Twisdale and are referred to as the "linear" and "uniform" cases. In developing these two cases, the Fujita windspeeds were used as a prior and were updated from data which give upper-limit values of the windspeed ranges. Failure calculations were performed and limiting windspeed bounds were calculated. For the uniform case the windspeed range was assumed to be uniform, while for the linear case, it was assumed to vary linearly from the upper bound to a zero likelihood at the lower bound.

Tornado Windspeed Frequency Analysis

The equations for a tornado hazard analysis, based on a single set of model assumptions, for the point strike and lifeline terms are given by the following equations:

$$\text{Point Strike: } F(V) = OR[CVLVW][AI \cdot OI] \quad (3.1)$$

$$\text{Lifeline: } F(V) = OR[VL][LI \cdot OI]A \quad (3.2)$$

where: $F(V_j)$ = Frequency of occurrence of tornadoes with a maximum windspeed equal to V_j . $F(V)$ is the vector of values for the seven windspeeds.

OR = Occurrence rate of tornadoes (see Table 3-1)

$[CVLVW]$ = Combined area variation matrix (see Table 3-2)

$[AI]$ = Average tornado area vector (see Table 3-1)

$[OI]$ = F-scale occurrence fraction vector (see Table 3-1)

$[VL]$ = Damage length variation matrix (see Table 3-3)

$[LI]$ = Average tornado length vector (see Table 3-1)

A = Width of plant site (see Figure 3-2b)

The wind hazard curve is just the frequency of exceeding a windspeed V_j as given by the following equation:

$$F(V_j) = \sum_{i=j}^n F(V_i) \quad (3.3)$$

Figure 3-4 shows an example analysis in expanded matrix form. The example shown corresponds to the top branches in the analysis flowchart. The last column contains the frequencies of occurrence corresponding to the windspeeds given in the first column. By successively accumulating the frequencies from the bottom of the last column, the hazard curve values are obtained (i.e., as given by Equation 3.3).

3.1.2 Straight Winds

The uncertainty analysis for straight winds is based on the assumption that the windspeed data recorded at Little Rock, Arkansas fits a Type I extreme value distribution (Ref. 10). This is a commonly accepted assumption for straight winds and is the same distribution that the USNRC used to develop the mean hazard curve for straight winds. The error in the hazard based on the recorded data is a consequence of the limited size of the data sample. The procedure based on the method of moments was used to obtain the estimated standard deviation of the sampling error (Refs. 6 and 7). The distribution of windspeeds at each frequency of exceedance was considered to be normally distributed. The distribution of windspeeds is relatively tight, thus the normal assumption is reasonable.

The uncertainty due to the proximity of the weather station to the site and the differences in terrain roughness between the site and the station were considered. If a station is located far from a site, then other adjacent stations should also be included in the analysis. In addition, if the terrain roughness is different then a windspeed correction factor, which is uncertain, should be included.

For ANO-1, the closest weather station is only 57 miles away and the terrain in between is similar. Thus, no uncertainties for station distance or differences in terrain roughness were considered. Thus, the entire uncertainty for ANO-1 straight winds was based on the error due to the limited sample given in Reference 6.

The Type I distribution which was used to determine the uncertainty is given by the following equation:

$$F(V) = \exp \left[-\exp \left(-\frac{V-\mu}{\sigma} \right) \right] \quad (3.4)$$

where μ and σ are the location and scale parameters, respectively. From the weather station data, the mean, $\bar{X}$, and sample standard deviation, S , are obtained and the location and scale parameters are found by the following relationships:

$$\bar{X} = \mu + 0.5772 \quad (3.5)$$

$$S = \frac{\pi \sigma}{\sqrt{6}} \quad (3.6)$$

The estimated standard deviation of the windspeed sampling error, SD , is given by Equation 3.7.

$$SD = \left[\frac{\pi^2}{6} + \frac{1.1396(y-0.5772)}{\sqrt{6}} + 1.1(y-0.5772)^2 \right]^{1/2} \frac{S\sqrt{6}}{\pi\sqrt{n}} \quad (3.7)$$

where: $y = -\ln[-\ln(F(V))]$
 n = Number of data points

3.2 Results

As indicated by Figure 3-1 a total of 72 hazard curves were run for the tornado analysis with corresponding probability values which varied from 0.0001 to 0.14. The probability distribution on the frequency of exceedance was shifted at each frequency so that the mean frequency values were equal to those provided by the USNRC. By synthesis, 5 hazard curves, each with equal probability values (i.e., 0.20) were calculated. Table 3-5 gives the hazard data including the mean frequency values and Figure 3-5 shows a plot of the 5 hazard curves.

TABLE 3-1
TORNADO ANALYSIS DATA

F-Scale	Regional Data			Local Data		
	Occurrence Fraction [OI]	Average Area (mi ²) [AI]	Average Length (mi) [LI]	Occurrence Fraction [OI]	Average Area (mi ²) [AI]	Average Length (mi) [LI]
F-0	.1658	.30	1.53	.1356	.05	.76
F-1	.3379	.72	3.01	.3299	.32	3.12
F-2	.3122	1.8	4.76	.3368	1.22	7.23
F-3	.1322	4.3	9.15	.1473	2.94	13.33
F-4	.0413	8.5	18.8	.0426	5.40	24.63
F-5	.0093	15.7	26.9	.007	15.7	26.9
F-6	.0013	18.9	30.1	.0008	18.9	30.1

Rate of Occurrence, OR: 5.179-4/mi²/yr

4.372-4/mi²/yr

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TABLE 3-2
COMBINED AREA VARIATION MATRIX, [CVCVW]

Reinhold 1 (Ref. 9)

Local Tornado State	True Maximum Tornado State						
	F-0	F-1	F-2	F-3	F-4	F-5	F-6
F-0	1.000	.743	.658	.615	.637	.632	.625
F-1	0	.257	.248	.267	.234	.236	.238
F-2	0	0	.094	.091	.093	.088	.089
F-3	0	0	0	.027	.028	.033	.033
F-4	0	0	0	0	.008	.009	.011
F-5	0	0	0	0	0	.002	.003
F-6	0	0	0	0	0	0	.001

Reinhold 2 (Ref. 9)

Local Tornado State	True Maximum Tornado State						
	F-0	F-1	F-2	F-3	F-4	F-5	F-6
F-0	1.000	.751	.484	.316	.315	.293	.289
F-1	0	.249	.414	.423	.371	.330	.326
F-2	0	0	.102	.205	.199	.201	.176
F-3	0	0	0	.056	.089	.112	.107
F-4	0	0	0	0	.026	.050	.063
F-5	0	0	0	0	0	.014	.030
F-6	0	0	0	0	0	0	.009

WASH-1300 (Ref. 4)

1.0 along diagonal; zero elsewhere

TABLE 3-3

DAMAGE LENGTH VARIATION MATRIX, [VL]

Reinhold 1 (Ref. 9)

Local Tornado State	Recorded Tornado State						
	F-0	F-1	F-2	F-3	F-4	F-5	F-6
F-0	1.000	.383	.180	.077	.130	.118	.100
F-1	0	.617	.279	.245	.131	.125	.110
F-2	0	0	.541	.310	.248	.162	.120
F-3	0	0	0	.368	.234	.236	.160
F-4	0	0	0	0	.257	.187	.200
F-5	0	0	0	0	0	.172	.150
F-6	0	0	0	0	0	0	.160

Reinhold 2 (Ref. 9)

Local Tornado State	Recorded Tornado State						
	F-0	F-1	F-2	F-3	F-4	F-5	F-6
F-0	1.000	.572	.280	.116	.142	.133	.120
F-1	0	.428	.352	.245	.158	.102	.130
F-2	0	0	.368	.318	.278	.189	.150
F-3	0	0	0	.321	.210	.242	.165
F-4	0	0	0	0	.212	.185	.175
F-5	0	0	0	0	0	.149	.140
F-6	0	0	0	0	0	0	.120

WASH-1300 (Ref. 4)

1.0 along diagonal; zero elsewhere

TABLE 3-4
F-SCALE WINDSPEED CONVERSION

F-Scale	Fujita	Windspeed (mph)	
		Twisdale (Ref. 8)	
		Linear	Uniform
F-0	40-72	40-73	40-65
F-1	73-112	73-103	65-96
F-2	113-157	103-135	96-114
F-3	158-206	135-168	134-139
F-4	207-260	168-209	139-181
F-5	261-318	209-277	181-236
F-6	319-380	295*	289*

* Mean value extrapolated

TABLE 3-5
TORNADO HAZARD DATA

Windspeed (mph)	Mean Value*	Frequency of Exceedance Per Year				
		Probability Value				
		0.20	0.20	0.20	0.20	0.20
66	1.53-3	9.66-4	1.10-3	1.48-3	1.75-3	2.35-3
93	8.33-4	3.69-4	5.58-4	7.61-4	9.83-4	1.49-3
134	2.81-4	4.86-5	1.07-4	1.91-4	3.09-4	7.49-4
182	5.61-5	2.55-6	9.84-6	2.15-5	4.81-5	1.98-4
234	1.03-5	3.54-7	7.14-7	1.89-6	6.18-6	4.25-5
290	2.00-6	4.38-9	4.57-8	1.51-7	5.83-7	9.20-6
349	2.07-7	1.49-10	2.10-9	8.88-9	4.24-8	9.82-7
Any Size Tornado	1.53-3	9.66-4	1.10-3	1.48-3	1.75-3	2.35-3

* Equal to the USNRC best estimate values for ANO-1

TABLE 3-6
STRAIGHT WIND HAZARD DATA

Windspeed (mph)	Mean Value*	Frequency of Exceedance Per Year				
		Probability Value				
		0.20	0.20	0.20	0.20	0.20
79	1.00-2	7.13-4	3.74-3	7.47-3	1.29-2	2.52-2
84	5.00-3	2.38-4	1.55-3	3.40-3	6.30-3	1.35-2
91	2.00-3	5.32-5	4.74-4	1.18-3	2.42-3	5.87-3
97	1.00-3	1.60-5	1.86-4	5.17-4	1.16-3	3.12-3
102	5.00-4	5.22-6	7.54-5	2.30-4	5.53-4	1.64-3
109	2.00-4	1.14-6	2.24-5	7.77-5	2.07-4	6.92-4
115	1.00-4	3.36-7	8.60-6	3.34-5	9.67-5	3.61-4
128	2.00-5	2.11-8	9.58-7	4.73-6	1.65-5	7.78-5
133	1.00-5	6.72-9	3.81-7	2.06-6	7.73-6	3.98-5
146	2.00-6	4.12-10	4.14-8	2.85-7	1.29-6	8.38-6
151	1.00-6	1.30-10	1.63-8	1.23-7	6.00-7	4.26-6

* Equal to USNRC best estimate values for ANO-1

Tornado
Plant Strike Model

Plant Site Width

Tornado Analysis
Data

Tornado Damage
Area and Length
Variation Matrices

F-Scale
Windspeed
Conversion

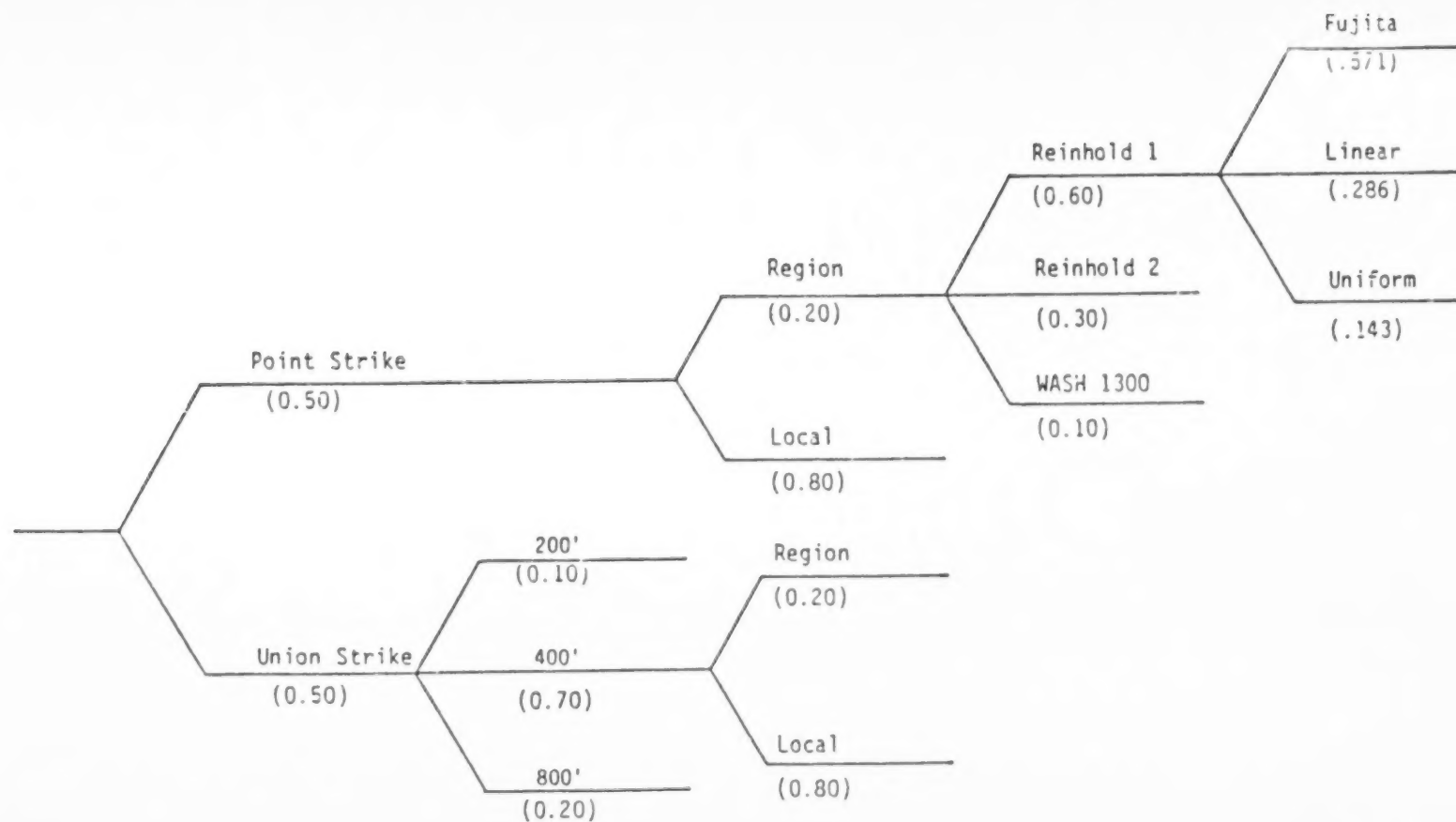
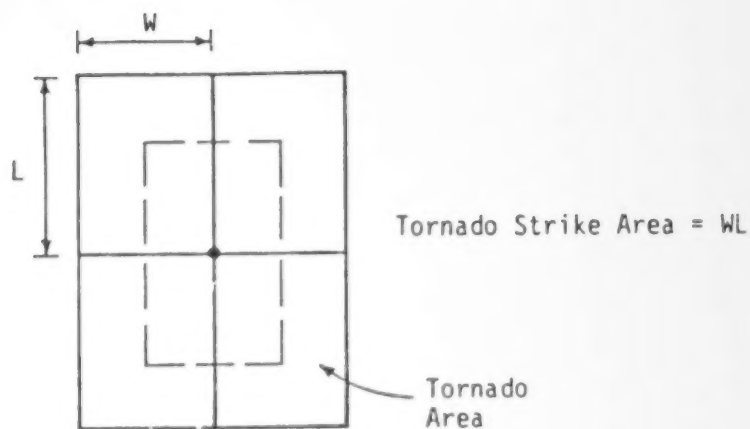
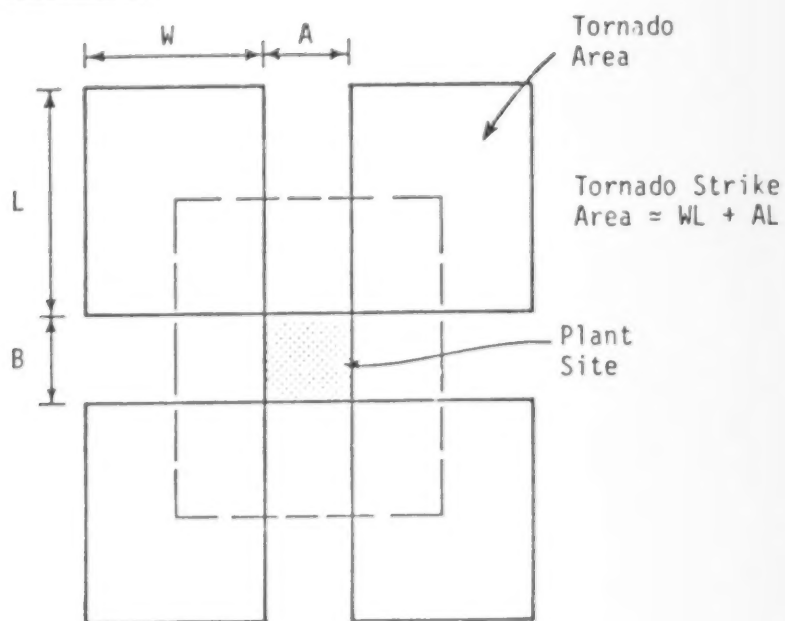


FIGURE 3-1 UNCERTAINTY ANALYSIS FLOWCHART

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(a) Point Strike



(b) Union Strike

FIGURE 3-2 TORNADO STRIKE MODELS

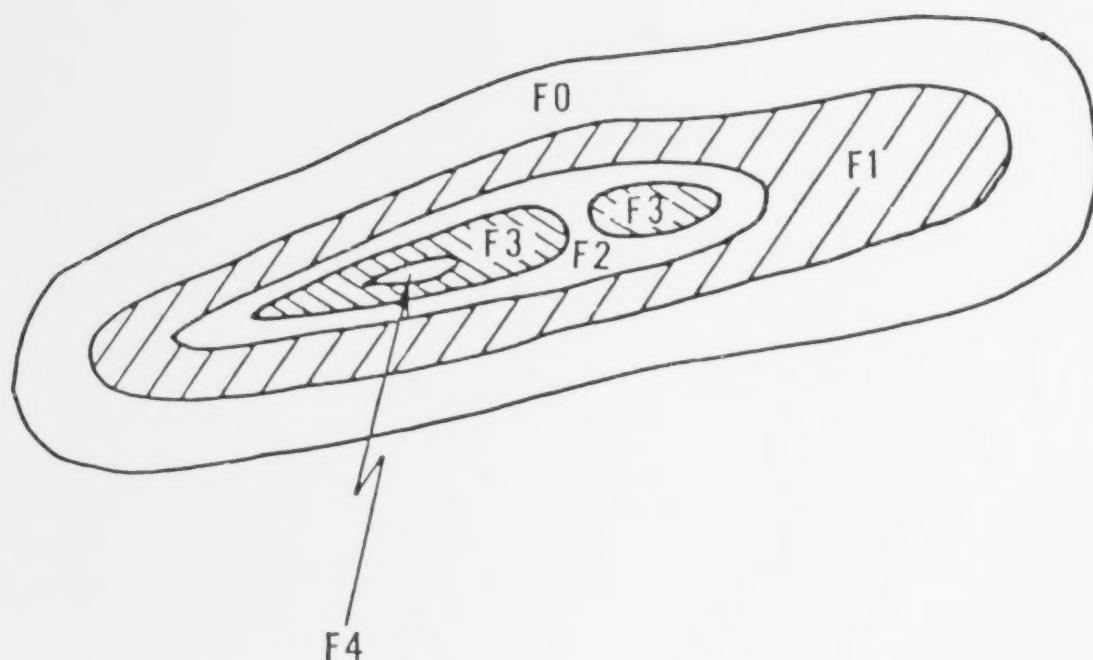


FIGURE 3-3 SKETCH OF HYPOTHETICAL F-4 TORNADO ILLUSTRATING VARIATION OF INTENSITY (Ref. 9)

$$\begin{array}{c}
 F \begin{bmatrix} 56 \text{ mph} \\ 93 \\ 134 \\ 182 \\ 234 \\ 290 \\ 349 \end{bmatrix} = F \begin{bmatrix} F-0 \\ F-1 \\ F-2 \\ F-3 \\ F-4 \\ F-5 \\ F-6 \end{bmatrix} = 5.179-4 \text{ Per Year}
 \end{array}
 \begin{array}{c}
 \begin{bmatrix} 1.000 & .743 & .658 & .615 & .637 & .632 & .625 \\ 0 & .257 & .248 & .267 & .234 & .236 & .238 \\ 0 & 0 & .094 & .091 & .093 & .088 & .089 \\ 0 & 0 & 0 & .027 & .028 & .033 & .033 \\ 0 & 0 & 0 & 0 & .008 & .009 & .011 \\ 0 & 0 & 0 & 0 & 0 & .002 & .003 \\ 0 & 0 & 0 & 0 & 0 & 0 & .001 \end{bmatrix}
 \end{array}
 \begin{array}{c}
 \begin{bmatrix} .30 & .1658 \\ .72 & .3379 \\ 1.8 & .3122 \\ 4.3 & .1322 \\ 8.5 & .0413 \\ 15.7 & .0093 \\ 18.9 & .0013 \end{bmatrix}
 \end{array}
 = \begin{array}{c}
 \begin{bmatrix} 6.63-4 \\ 2.47-4 \\ 7.88-5 \\ 1.60-5 \\ 2.27-6 \\ 1.89-7 \\ 1.27-8 \end{bmatrix}
 \end{array}
 \begin{array}{c}
 \text{Per} \\ \text{Year}
 \end{array}$$

[OR] [CVLVW] [AI - 01] F(V)

FIGURE 3-4 EXAMPLE ANALYSIS FOR TOP BRANCHES ON FIGURE 3-1

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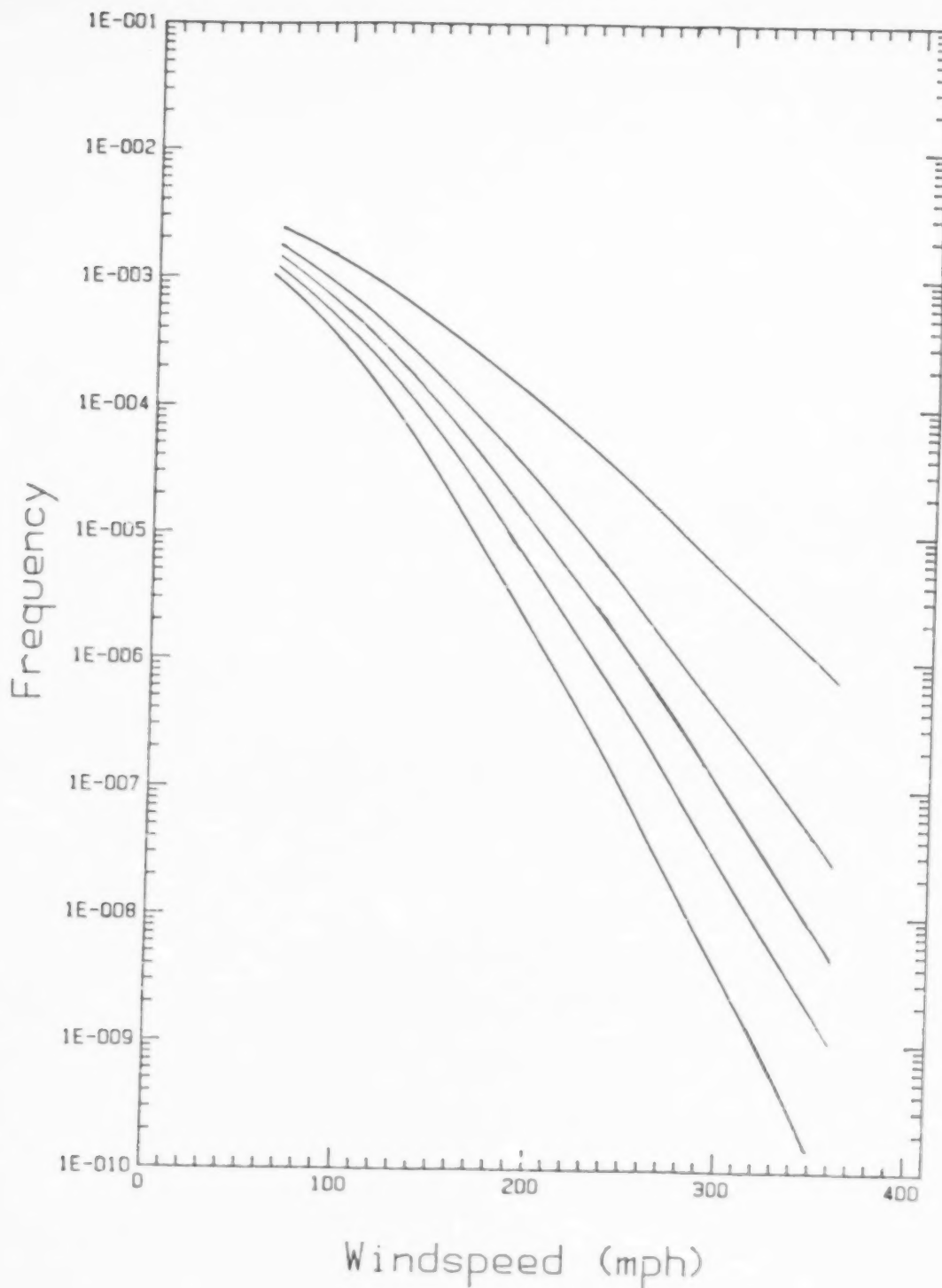


FIGURE 3-5 TORNADO HAZARD CURVES*

* Five possible hazard curves; each with a 0.2 probability of occurrence.

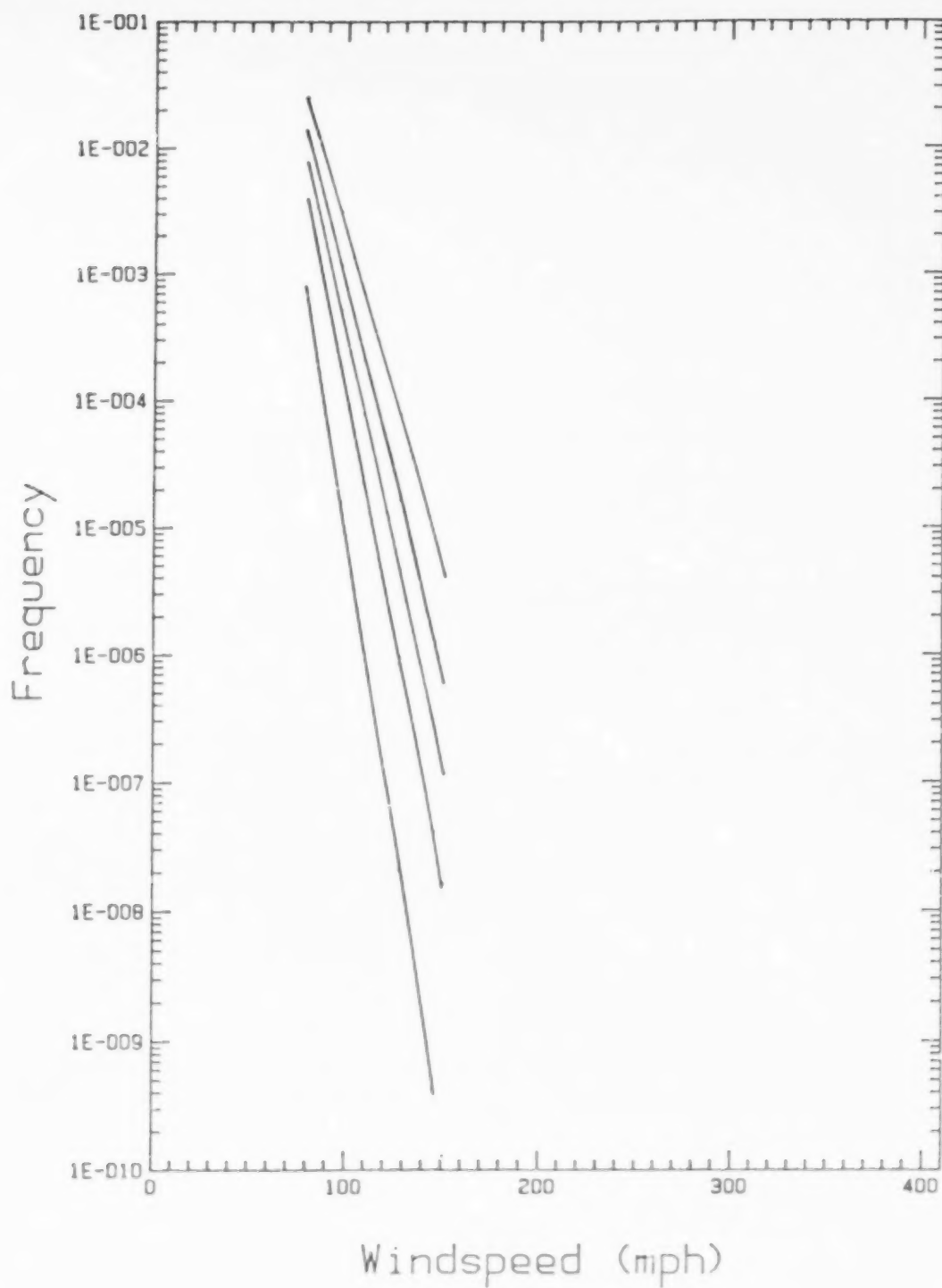


FIGURE 3-6 STRAIGHT WIND HAZARD CURVES*

* Five possible hazard curves; each with a 0.2 probability of occurrence.

4.0 TORNADO MISSILE ANALYSIS

A tornado missile analysis was performed to develop the probability distribution on the conditional fraction of missile impact on exposed equipment, which could fail due to missile penetration.

The process of missile impact starts with the injection of a missile into the wind stream which is carried through the air where it could impact exposed equipment or a structure which contains safety-related equipment. The damage caused by a missile is a function of its mass, terminal velocity, deformation characteristics, and angle of impact. The last variable is extremely important since a head-on impact is more damaging than a glancing or side blow.

For structures which contain equipment, several additional events are required to complete the impact scenario. The missile must either penetrate the structure and travel further and impact the equipment, or in the case of concrete, the backside of the wall must scab and a piece of concrete fly through the air and strike the equipment. At each step the probability that a missile (or scabbed piece of concrete) impacts the equipment is less than one, and in simplified terms the failure fraction is the product of individual fractions of occurrence corresponding to the events described above.

The methodology used to perform the tornado missile analysis is given in Section 4.1 and the results of the analysis for ANO-1 is presented in Section 4.2.

4.1 Methodology

A simplified tornado missile analysis procedure was developed similar to the analysis conducted for the Oconee PRA (Ref. 11). A normalized tornado missile impact parameter, ψ , was defined as follows:

$$\psi = \text{Frequency of impact/missile/unit target area/tornado point strike frequency}$$

From past tornado missile impact studies, values of ψ were calculated. It was postulated that the normalized tornado missile impact parameter depends on the following variables:

- Size of target
- Relative F-scale distribution in the region surrounding the site
- Type and location of missiles
- Arrangement of buildings and location of missile targets

Tornado simulation studies using the TORMIS computer code developed by EPRI (Refs. 5 and 12) were used as the basis for calibrating the simplified analysis. Reference 12 gives missile impact values for two plants (A and B1) which were used to develop ψ -values for large targets (i.e., buildings, tanks, etc.). When normalizing the data for the two plants by USNRC Regions I, II, and III, it was found that the results for the three regions were within a factor of 1.5 with even a smaller difference between Regions I and II. This indicates that the relative F-scale distribution between regions does not have a significant effect on the ψ -value. Parameter values were calculated for all the buildings at the two plants using the slightly higher Region I data and combined into a common distribution; thus, uncertainty in the type and location of missiles and arrangement of buildings was reflected in the variability of the final results.

Table 4-1a gives the resulting normalized tornado missile impact parameter values for large structures in USNRC Regions I and II for 5 equal probabilities. The values range over an order of magnitude with a mean value equal to $1.23 \cdot 10 / \text{missile} / \text{ft}^2$ of target area/tornado point strike frequency. The values of ψ are normalized with respect to a point strike frequency since this is the basis that the USNRC hazard analysis used. The values in Table 4-1a are applicable for large structures and tanks.

A tornado missile analysis conducted for the Seabrook Station developed impact values for small targets approximately 100 to 1000 ft² in size (Ref. 13). Normalized tornado missile impact parameters were developed for each of the targets and the values were found to range from $7.44 \cdot 13$ to $1.17 \cdot 8 / \text{missile} / \text{ft}^2$ of target area/tornado point strike frequency. This range covers over four orders of magnitude. However, the analysis showed that the exposure of the target relative to missile location has a significant effect on the normalized tornado missile impact parameter. In essence, the large structure values given in Table 4-1a represent average values from small areas with some ψ -values being large and others small. Thus the variability for large structures was not found to be as large as for small target areas.

The 30 target areas in the Seabrook study were segregated into high, medium and low exposure areas. The process of grouping the targets was based on judgment and considered where the targets were located relative to the concentration of missiles and shielding effects of adjacent buildings. Table 4-1b summarizes the results that were found. The variability of ψ -values for each of the three categories is reduced from the case where all data are considered in one group and is comparable with the variability found for the large structure data. In addition, the large structure values were found to be slightly higher than the medium exposure values for small target areas, which is consistent.

The ψ -values in Table 4-1 were used to determine the likelihood of missile impact conditional on tornado occurrence. The equipment target areas were conservatively determined and considered to be known with certainty and are given in Table 4-2. For each of the targets a conservative distribution on exposure was assumed to be 0.1, 0.8 and 0.1 for high, medium and low exposures. Also, the number of tornado missiles was conservatively estimated to be 5,000, 20,000, and 60,000 with corresponding probability weights of 0.2, 0.6 and 0.2. This distribution of missile inventory is based on judgment and comparisons of the general missile environment at ANO-1 to more accurate counts of missiles at other nuclear power plants. The diesel generator exhaust stacks were analyzed using the small target area data, while the conditional fractions for the BWSTs, CSTs and the diesel fuel oil storage tank are based on the large target area data in Table 4-1.

4.2 Results

Table 4-2 gives the fraction of failure for missile impact conditional on the occurrence of a tornado. Five equal probability values (i.e., 0.20) and the mean value are given for each component. The frequency of occurrence of tornadoes is given in the last row of Table 3-5.

The failure fractions in Table 4-2 can be considered to be independent. For systems in parallel, detailed tornado missile impact studies show that for two components in parallel the conditional fractions can be multiplied together if the reference tornado frequency of occurrence corresponds to a point strike frequency. The tornado occurrence frequency values given in the last row of Table 3-5 are based on a point strike frequency analysis; thus, the missile impact values given in Table 4-2 can be considered to be independent when used for parallel components in a systems analysis.

TABLE 4-1
NORMALIZED TORNADO MISSILE IMPACT
PARAMETER VALUES (ψ)

a) <u>Large Structures in USNRC Regions I and II</u>		
<u>Value</u>	<u>Probability</u>	<u>Normalized Parameter, ψ^*</u>
1	0.20	1.49-11
2	0.20	3.49-11
3	0.20	7.16-11
4	0.20	1.61-10
5	0.20	3.31-10
Mean		1.23-10

b) <u>Small Target Areas in USNRC Regions I and II</u>				
<u>Value</u>	<u>Probability</u>	<u>Normalized Parameter, ψ^* by Exposure</u>		
		<u>High</u>	<u>Medium</u>	<u>Low</u>
1	0.20	1.42-10	1.30-11	1.61-12
2	0.20	4.03-10	2.52-11	4.09-12
3	0.20	1.63-9	4.86-11	7.71-12
4	0.20	2.57-9	1.14-10	1.46-11
5	0.20	7.33-9	2.31-10	4.87-11
Mean		2.42-9	8.64-11	1.54-11

* Units are per missile per ft² of target area per tornado point strike frequency

TABLE 4-2
CONDITIONAL FRACTION OF FAILURE FOR MISSILE IMPACT

Component	Probability Weight					Mean Value	Target Area (ft ²)
	0.20	0.20	0.20	0.20	0.20		
Diesel Generator Exhaust Stacks (each stack)	1.80-3	1.48-4	5.30-5	2.25-5	7.96-6	4.06-4	52
New 321,000 Gallon CST**	7.78-4	2.22-3	4.65-3	1.09-2	3.36-2	1.05-2	3400
BWST*	8.69-4	2.48-3	5.20-3	1.22-2	3.75-2	1.17-2	3800
Existing CST*	7.54-4	2.15-3	4.52-3	1.06-2	3.26-2	1.01-2	3300
Diesel Fuel Oil Storage Tank*	7.09-4	2.02-3	4.24-3	9.95-3	3.06-2	9.51-3	3100

* Does not include roof, since missile impact would not affect supply of water or oil.

** Based on wall area above protective missile shield wall (5-feet high) and does not include roof. Note that the 5 foot-high lower portion of the new CST contains 50,000 gallons which is protected against missiles.

5.0 WIND PRESSURE FRAGILITY ANALYSIS

Tornado and straight wind pressures can cause failure of weak structures and exposed equipment. Fragility curves, which relate the failure fraction as a function of the windspeed, portray the capacity characteristics of structural components. The methodology used to perform the tornado missile analysis is given in Section 5.1 and the results of the analysis for ANO-1 are presented in Section 5.2.

5.1 Methodology

In a manner similar to the procedure used to develop fragility curves for the effects of earthquakes (Ref. 14), wind capacity is modeled as a product of random variables and is expressed in terms of windspeed. The fragility curve distribution is assumed to be lognormal. Thus the fraction of failure is a function of the following four parameters:

- Median windspeed capacity, $\check{V}$
- Logarithmic standard deviation for capacity due to randomness, β_r
- Logarithmic standard deviation of the median capacity due to uncertainty, β_u
- The windspeed, V

The process for calculating a family of fragility curves for a component can be best thought of as consisting of two steps. In the first step, the density function of the median wind capacity, which is lognormal $LN(\check{V}, \beta_u)$, can be divided into n equal median values. Each median value has equal probability corresponding to the n fragility curves desired for the component (i.e., $\check{V}_i$, $i = 1, n$). Then in the

second step, n fragility curves are calculated, where each curve comes from the lognormal distribution, $LN(\check{V}_i, \beta_r)$, which is evaluated at all wind velocity values of interest. Equation 5.1 gives the relationship for a single fragility curve.

$$F(V|\check{V}_i) = \Phi \left[\frac{\ln V / \check{V}_i}{\beta_r} \right] \quad (5.1)$$

where: $F(V|\check{V}_i)$ = The fraction of failure evaluated at wind velocity, V , given the median value, $\check{V}_i$

$\Phi[\cdot]$ = Standard cumulative normal distribution

Equation 5.1 is repeated for each of the n median values, $\check{V}_i$, to form the family of fragility curves, each of which has a probability value of $1/n$.

The development of the fragility parameter values for the potentially vulnerable components is based on approximate procedures which use the design wind capacity whenever possible as the starting point for the analysis. This allows a capacity estimate to be made without performing a detailed stress analysis. For some components such as diesel generator exhaust stacks, the original design capacities are not readily available. In the fragility analysis conservatisms inherent in the design process are factored out of the capacity to obtain a realistic estimate of the actual failure capacity.

The parameters which in general contribute to the capacity of a component for the effects of wind pressure are given in Table 5-1.

Because the large buildings at nuclear power plants tend to shield each other, windspeeds vary due to the effects of funneling (i.e., windspeed is increased as the air is channeled through a constricted

space) or blockage (i.e., windspeed is decreased as wind is slowed due to obstructions). These effects can have a large influence on the wind velocities which impinge on structures and equipment.

Wind pressure and forces are related to the wind velocity through the use of a shape coefficient. Values for typical structural shapes can be found in References 15 and 16. Concentrated pressures on small areas corresponding to local failure modes (e.g., failure at corner eaves of metal buildings) are generally larger than average pressures over an entire building.

For straight winds, the free-field windspeed varies vertically from the ground to the gradient wind elevation and is a function of the roughness of the ground terrain (Ref. 17). For straight wind, the vertical wind profile is considered in the analysis. In contrast, a uniform windspeed distribution vertically is assumed for tornado winds.

The development of the strength characteristics in the fragility analysis are similar to the development of the strength factor in a seismic fragility assessment. Similar information is required both for the seismic and wind pressure strength analysis, but energy absorption capacity utilized in the seismic fragility analysis is not appropriate for static wind loading. However, a small margin may exist for ductile elements between yield and ultimate capacity due to strain hardening. Special attention is directed to potential brittle failures such as buckling collapse or failure of welds.

5.2 Results

Table 5-2 lists the fragility parameter values for each diesel generator exhaust stack. Plots of the fragility curves are given in Figure 5-1.

The fragility curves for the exhaust stack are based on the capacity of the Class 2 portion of the auxiliary building, which is assumed to be designed for winds up to 67 mph. It is likely that the auxiliary building siding will fail at a higher windspeed. If the siding fails in the vicinity of the stack, it is not obvious that the stack will be damaged to the point that the exhaust fumes are blocked off. However, the fragility analysis assumes that this will occur. This situation could be rectified by an emergency operations procedure which would require that the two stacks be checked after a tornado or periodically during a severe storm. If a stack fails, it could be cut off below the collapsed section. An alternative would be to provide a blind flange near the bottom which could be removed if the failed section occurs above the blind flange.

Note that since the two exhaust stacks are located near each other and are both attached to the same metal siding, if one fails the other also will likely fail. Hence, failure of the diesel generator exhaust stacks should be considered to be perfectly dependent.

TABLE 5-1
WIND PRESSURE FRAGILITY PARAMETERS

Wind Pressure/Force Relationship

- Wind funneling and blockage around plant structures
- Pressure shape coefficient
- Wind pressure model
- Vertical distribution of wind velocity

Structural Strength Characteristics

- Strength model
- Failure stress level

TABLE 5-2
FRAGILITY PARAMETER VALUES

Component	Median Wind Velocity (mph)	β_r	β_u	β_c
Diesel Generator Exhaust Stacks (each stack)	140	0.10	0.32	0.34

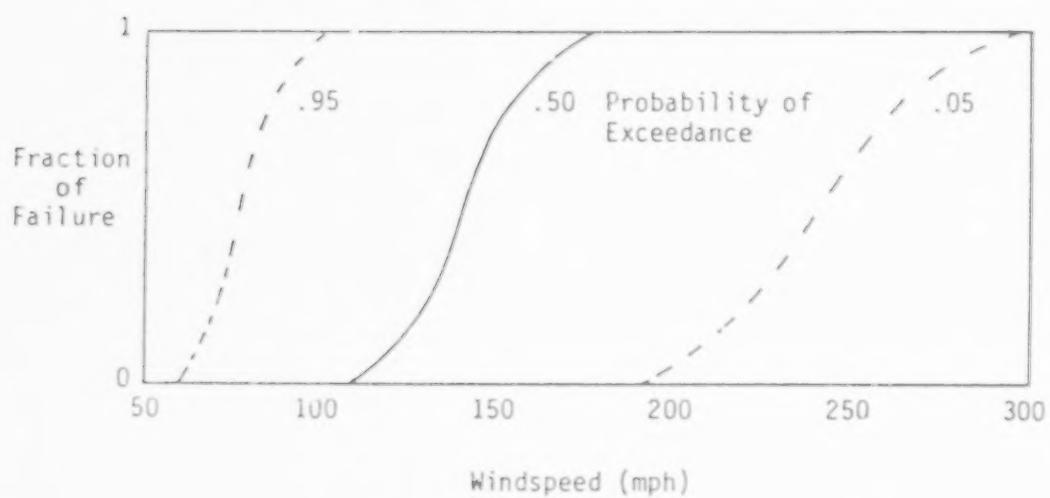


FIGURE 5-1 DIESEL GENERATOR EXHAUST STACK
FRAGILITY CURVES

6.0 SYSTEM ANALYSIS

Tornado missiles, tornado winds, and straight winds can cause failure of various systems in the plant. In this section the response of the plant to the various system failures is modeled. This model is used in Section 7.0 for quantifying the risk due to extreme winds.

The methodology used to perform the systems analysis is given in Section 6-1. The results of the analysis for ANO-1 are given in Section 6-2.

6.1 Methodology

To determine the contribution (if any) that tornado winds and missiles, and straight winds can have to core melt probability the following steps are taken:

- o System failures that will occur as a result of vulnerable equipment/structure failures are determined.
- o An event tree is constructed that models the plant's response to potential system failures assuming a simultaneous loss of offsite power (LOSP). Assuming that offsite power fails simultaneously is considered realistic for the severity of the storms being analyzed. The LOSP event tree in the internal systems analyses is used as a model in constructing the wind/tornado event tree.

6.2 Results

The results for the wind/tornado systems analysis are given in the following two subsections.

6.2.1 Component/System Failures Due to Winds/Tornadoes

The systems that will fail as a result of the failure of the various vulnerable equipment/structures are shown in Table 6-1. The vulnerable

TABLE 6-1
SYSTEMS THAT FAIL AS A RESULT OF THE FAILURE OF
VULNERABLE EQUIPMENT/STRUCTURES AT ANO-1

Vulnerable Structure/ Equipment	System(s) Failed ⁽¹⁾	Comments
Diesel Generator (DG) Exhaust Stacks	HPIS/LPIS/HPRS/LPRS/EFWS ⁽²⁾	Vulnerable to both wind pressure and tornado missiles
Diesel Fuel Oil Storage Tank (DFOST)	HPIS/LPIS/HPRS/LPRS/EFWS ⁽²⁾	Vulnerable to tornado missiles
Condensate Storage Tanks (CSTs)	EFWS	Vulnerable to tornado missiles
Borated Water Storage Tank (BWS)	HPIS/LPIS	Vulnerable to tornado missiles

- (1) HPIS = High Pressure Injection System
 LPIS = Low Pressure Injection System
 HPRS = High Pressure Recirculation System
 LPRS = Low Pressure Recirculation System
 EFWS = Emergency Feedwater System

- (2) The EFWS will also fail. The motor driven pump will fail due to LOSP and diesel generator failure. The turbine driven pump could fail due to battery depletion, random failures, test or maintenance, CST depletion, etc.

equipment structures listed in the left column are obtained from Table 2-1.

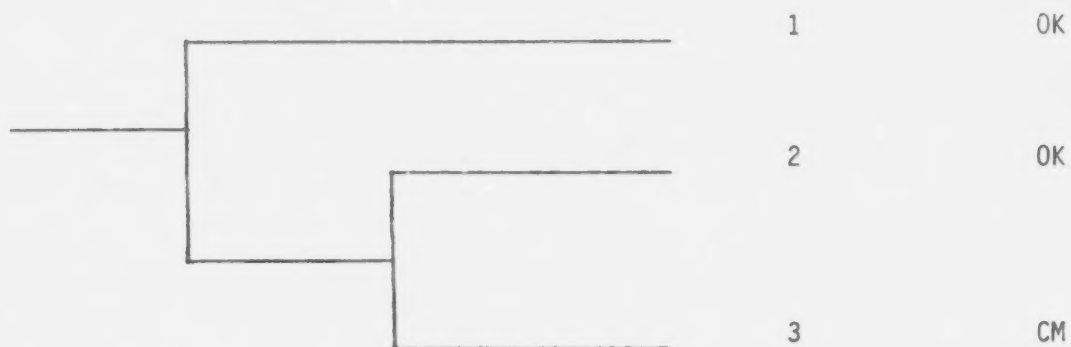
Each system (s) failure is discussed below:

1. HPIS/LPIS/HPRS/LPRS/EFWS Failure - The assumption is made that a simultaneous Loss of Offsite Power (LOSP) occurs during a wind/tornado initiated transient event. Therefore, failure of the diesel generators will result in the loss of all offsite and emergency onsite AC power. This will fail both early (HPIS/LPIS) and late (HPRS/LPRS) Emergency Coolant Injection (ECI) systems. The loss of the DGs will fail the motor driven pump train of the EFWS. The turbine driven pump train can fail either randomly or due to battery depletion (or due to T&M, or valves, etc).
2. HPIS/LPIS/HPRS/LPRS/EFWS Failure - Failure of the diesel fuel oil storage tanks will fail both diesel generators. This is the same as (1) above.
3. EFWS Failure - The CSTs provide the water supply to the EFWS pumps. Therefore, loss of the CSTs is assumed to cause EFWS failure.
4. HPIS/LPIS Failure - The BWST is the water source for HPIS and LPIS; therefore, failure of the BWST will cause failure of the early ECI systems.

6.2.2 Wind/Tornado Event Tree

A functional event tree models the response of ANO-1 to winds and/or tornadoes is shown in Figure 6-1. A simultaneous LOSP is assumed to occur. In Boolean equation form, the functions are systemically defined as the following:

Function	Secondary Cooling (SEC)	Emergency Coolant Injection (FNB mode)	Sequence Number	Core Status
Support Systems*	(CST, DFOST**, DGs**)	(BWST, DFOST, DGs)		



*BWST = Borated Water Storage Tanks
 DFOST = Diesel fuel oil storage tanks
 CST = Condensate Storage Tanks
 DGs = Diesel Generators

**The EFWS could fail either randomly or due to battery depletion eight hours after the DGs fail.

Figure 6-1. Wind/Tornado Functional Event Tree for ANO-1

Secondary Cooling = EFWS

where EFWS = Emergency Feedwater System

$$ECI = HPIS * LPIS + HPRS * LPRS$$

where ECI = Emergency Coolant Injection

HPIS = High Pressure Injection System (Feed and Bleed (FNB) mode)

*LPIS = Low Pressure Injection System

*HPRS = High Pressure Recirculation System

*LPRS = Low Pressure Recirculation System

Under each function in the event tree in Figure 6-1, the support systems that will cause the functions to fail are listed. The failure of the diesel generators (DGs) will fail the ECI system pumps and the EFWS MDP. Failure of the EFWS turbine train could occur either randomly or due to battery depletion in approximately 8 hours.

The sequence resulting in core melt (Sequence 3 in Figure 6-1) requires the failure of the secondary cooling and the emergency cooling injection systems. The probability of this sequence is evaluated in the following section.

*These systems would only be needed for long term cooling. The only part of ECI that is included in this analysis is the feed and bleed mode using HPIS.

7.0 RISK QUANTIFICATION

Tornado missiles, tornado winds, and straight winds can fail various plant systems resulting in a contribution to core melt probability. In this section the system failure probabilities due to winds and/or missiles are calculated. These system failure probabilities are used in the functional event tree developed previously to determine the contribution to core melt probability.

The methodology used to perform the risk quantification analysis is given in Section 7.1. The results of the analysis for ANO-1 given in Section 7.2.

7.1 Methodology

The total core melt probability is obtained in the following three steps:

- (1) The core melt probability contribution due to straight winds (SW) is calculated, and
- (2) The core melt probability due to tornado winds (TW) and tornado missiles (TM) is calculated, and
- (3) The total core melt probability is obtained by adding the results from steps (1) and (2) together.

7.1.1. Core Melt Probability Contribution Due to Straight Winds

The core melt probability due to straight winds is obtained in the following steps:

- o The probability of failure at each wind speed (fragility) is calculated for each function given in the event tree, Figure 6-1.
- o Using the above fragilities the probability of each sequence is determined for each wind speed.

- o The frequency of each sequence is obtained by multiplying the probability of each sequence for a given wind speed by the frequency of occurrence of that wind speed. The total sequence probability is obtained by adding the sequence probabilities at different wind speeds together.
- o The total probability of core melt is obtained by adding the total sequence frequencies together.
- o Recovery actions for each sequence are included whenever appropriate. A new core melt probability (with recovery) is calculated.

7.1.2 Core Melt Probability Contribution Due to Tornado Winds and Tornado Missiles

The same steps used for straight winds in Section 7.1.1 are followed.

7.1.3 Total Core Melt Probability

The total core melt probability is obtained by adding the contributions calculated for straight winds to that calculated for tornado winds and missiles.

7.2 Results

The core melt probabilities are given for straight winds, tornado winds and missiles, and the total (straight winds, tornado winds, and tornado missiles).

7.2.1 Straight Wind Results

The straight wind hazard/fragility for equipment/structures, systems and functions is given in Table 7-1. The contribution to core melt probability is given in Table 7-2 for the core melt sequence 3 in Figure 6-1.

The equipment/structure failure probabilities (fragility) are obtained from the fragility parameter values shown in Table 5-2. The hazard values are obtained from Table 3-6. The failure probability of both diesel generators

TABLE 7-1
STRAIGHT WIND HAZARD/FRAGILITY FOR
EQUIPMENT/STRUCTURES, SYSTEMS AND FUNCTIONS FOR ANO-1

Functional Failures		ECI (FNB mode/Secondary Cooling)
Equipment/Structural System Failures		Both Diesel Generators (Failure of both diesel generators exhaust stacks)
MPH	Frequency of exceedance	Failure Probability
79	1.00E-2	0
84	5.00E-3	0
91	2.00E-3	0
97	1.00E-3	0
102	5.00E-4	0
109	2.00E-4	0
115	1.00E-4	0
128	2.00E-5	0
133	1.00E-5	0
140	5.69E-6	1
146	2.00E-6	↓
151	1.00E-6	

TABLE 7-2
CORE MELT PROBABILITY DUE TO STRAIGHT WINDS, TORNADO
WINDS AND MISSILES, AND THE TOTAL COMBINATION

Sequence Number	Straight Winds		Tornado Winds and Missiles		Straight Winds, and Tornado Winds and Missiles	
	W/O Recovery	W/ Recovery	W/O Recovery	W/ Recovery	W/O Recovery	W/ Recovery
3	5.69E-6	1.16E-7	2.53E-4	5.19E-6	2.59E-4	5.31E-6

(DGs) is unity at 140 mph. This requires that both DG exhaust stacks fail. The probability that both stacks fail is simply the product of their two failure probabilities. In this case each failure probability is unity.

The core melt probability due to straight winds without recovery is $5.69E-6$. This core melt probability is due solely to the failure of the DGs at 140 mph. If the battery operated EFW turbine driven pump starts and runs successfully, this is a long-term sequence since steam generator dryout will not begin until after battery depletion occurs in approximately 8 hours. It will take an additional 4 hours until steam generator dryout occurs. This is longer than the 30 minutes needed for a transient without the EFW system at the onset of the accident due to the reduced heat level after 8 hours. The nonrecovery probability of offsite power $p(\text{LOSP}_{\text{NR}})$ is 0.03 in 12 hours. In addition to LOSP recovery there is a possibility of recovery by having someone cut off the bent portions of the DG exhaust stacks (caused by the straight winds) within the 12 hours. This is given a nonrecovery probability of 0.1. The incorporation of recovery is shown below for the loss of DGs when the turbine driven pump successfully operates.

$$\begin{aligned} p(\text{DGs fail}) * p(\text{LOSP}_{\text{NR}}) * p(\text{DG}_{\text{NR,MANACT}}) &= (1) (0.03) (0.1) \\ &= 3E-3 \end{aligned}$$

where $p(\text{DGs fail})$ = probability both DGs fail due to straight winds at 140 mph = 1.0

$p(\text{LOSP}_{\text{NR}})$ = LOSP nonrecovery probability in 12 hours = 0.03

$p(\text{DG}_{\text{NR,MANACT}})$ = nonrecovery probability that the DG exhaust stacks will be cut off if bent closed by winds = 0.1

If the EFW turbine driven pump randomly fails to start, core melt will occur after approximately 30 minutes. The probability of nonrecovery of offsite power in 30 minutes is 0.5 and it is not considered possible that the DG exhaust stacks could be fixed in this time. Failure of the turbine driven EFW pump to

start on demand is estimated to be $3.5E-2/\text{demand}$. Test and maintenance outage is a negligible contribution to EFW turbine pump failure. The incorporation of recovery is shown below for the loss of the DGs when the turbine driven pump has failed to start.

$$p(\text{DGs fail}) * p(\text{TDP fails}) * p(\text{LOSP}_{\text{NR}}) * p(\text{DG}_{\text{NR,MANACT}}) = (1)(3.5E-2)(.5)(1) = 1.74E-2$$

where $p(\text{DGs fail})$ = probability both DGs fail due to straight winds at 140 mph = 1.0

$p(\text{TDP fails})$ = probability that the turbine driven pump fails to start = $3.5E-2$

$p(\text{LOSP}_{\text{NR}})$ = LOSP nonrecovery probability in 30 minutes = 0.5

$p(\text{DG}_{\text{NR,MANACT}})$ = nonrecovery probability by cutting off the bent DG exhaust stacks = 1.0

The contribution to core melt due to straight winds is

$$(5.69E-6) * (3E-3 + 1.74E-2) = 1.16E-7$$

7.2.2 Tornado Winds and Missiles Results

The tornado winds and missiles hazard/fragility for equipment/structures, systems and functions is given in Table 7-3. The contribution to core melt probability is given in Table 7-2.

The hazard values are obtained from Table 3-5. The equipment/structure failure probabilities are obtained from the fragility parameter values in Table 5-2 for tornado winds and Table 4-2 for tornado missiles. At each wind velocity the fragility values are the probability failures due to both tornado winds and missiles. The probabilities are added in the Boolean manner:

TABLE 7-3
TORNADO WIND AND MISSILE HAZARD/FRAGILITY
FOR EQUIPMENT/STRUCTURES, SYSTEMS AND
FUNCTIONS FOR ANO-1

Failure Probabilities

Functional Failures		ECI**/Second- ary Cooling	ECI/Second- ary Cooling	Secondary Cooling	ECI
Equipment/Structures/ System Failures		DFOST**	Both Diesel Generators	CST**	BWST**
MPH	Frequency*	Failure Probabilities			
66	1.53E-3	9.51E-3(1)	1.65E-7(1)	1.0E-2(1)	1.17E-2(1)
93	8.33E-4	↓	1.65E-7	↓	↓
134	2.81E-4		1.65E-7		
140	2.53E-4		(2)		
182	5.61E-5		↓		
234	1.03E-5		↓		
290	2.00E-6		↓		
349	2.07E-7		↓		

* Frequency of exceedance

** ECI = Emergency Coolant Injection (Feed and Bleed (FNB) Mode)

CST = Condensate Storage Tank

BWST = Borated Water Storage Tank

DFOST = Diesel Fuel Oil Storage Tank

(1) Probability of missile impact

(2) Failure probability due to tornado winds

$$P_f(TM \text{ or } TW) = P_f(TM) + P_f(TW) - P_f(TM) \times P_f(TW)$$

where $P_f(TM \text{ or } TW)$ = probability of failure due to both tornado winds (TW) and missiles (TM)

$P_f(TM)$ = probability of failure due to TM

$P_f(TW)$ = probability of failure due to TW

In each case, the failure probabilities are due solely to tornado missiles until the wind velocities increase in magnitude to the point at which failures occur due to tornado winds as well as missiles.

To obtain the failure probabilities due to missiles for each equipment/structure, if a structure can fail due to various missile impacts, the total probability of failure is obtained by adding separate missile impact failure probabilities.

The failure probabilities of two different pieces of equipment/structures or systems is simply the product of their failure probabilities. For example, the failure probability of both diesel generators is unity at 140 mph, since the failure probability of each diesel generator stack is unity.

The core melt probability due to tornado winds and missiles is $2.53E-4$ if no recovery actions are considered. This core melt probability is due to failure of the DGs by tornado winds at 140 mph. It should be noticed that failure of the diesel fuel oil tank due to tornado missiles is not included since at ANO-1 there are two underground storage tanks plus two diesel day tanks. There is at least a seven day total diesel oil inventory available on-site in the emergency storage tanks for diesel generator operation during complete loss of electric power conditions.

The case where just the DGs fail is a long term sequence if the turbine driven EFW pump successfully runs, since steam generator dryout will not begin until after battery depletion occurs in approximately 8 hours, at which time the steam driven EFWS is assumed to fail. It will take an additional 4 hours until steam generator dryout occurs. This is longer than the 30 minutes needed for a transient without the EFW system at the onset of the accident. The nonrecovery

of offsite power $p(\text{LOSP}_{\text{NR}})$ is 0.03 in 12 hours. In addition to LOSP recovery, there is a possibility of recovery in this case by having someone cut off the bent portions of the DG exhaust stacks (caused by tornado winds) within 12 hours. This is given a nonrecovery probability of 0.1.

The incorporation of recovery is shown below for the loss of the DGs:

$$p(\text{DGs fail}) * p(\text{LOSP}_{\text{NR}}) * p(\text{DG}_{\text{NR}}, \text{MANACT}) = (1)(0.03)(0.1) \\ = 3\text{E-}3$$

where $p(\text{DGs fail})$ = probability both DGs fail due to tornado winds at 140 mph without recovery = 1.0

$$p(\text{LOSP}_{\text{NR}}) = \text{LOSP nonrecovery in 12 hours} = 0.03$$

$p(\text{DG}_{\text{NR}}, \text{MANACT})$ = nonrecovery probability that the DG exhaust stacks will be cut off if bent closed by winds = 0.1

If the EFW turbine driven pump randomly fails to start, core melt will occur after approximately 30 minutes. The probability of nonrecovery of offsite power in 30 minutes is 0.5 and it is not considered possible that the DG exhaust stacks could be fixed in this time. Failure of the turbine driven EFW pump to start on demand is estimated to be $3.5\text{E-}2/\text{demand}$. Test and maintenance outage is a negligible contributor. The incorporation of recovery is shown below for the loss of the DGs when the turbine pump has failed to start.

$$p(\text{DGs fail}) * p(\text{TDP fails}) * p(\text{LOSP}_{\text{NR}}) * p(\text{DG}_{\text{NR}}, \text{MANACT}) = (1)(3.5\text{E-}2)(.5)(1) = \\ 1.75\text{E-}2$$

where $p(\text{DGs fail})$ = probability both DGs fail due to straight winds at 140 mph = 1.0

$p(\text{TDP fails})$ = probability that the turbine driven pump fails to start = $3.5\text{E-}2$

$$p(\text{LOSP}_{\text{NR}}) = \text{LOSP nonrecovery probability in 30 minutes} = 0.5$$

$p(DG_{NR}, MANACT)$ = nonrecovery probability by cutting off the bent DG exhaust stacks = 1.0

The contribution to core melt frequency due to diesel generator exhaust stack failure with recovery is:

$$(2.53E-4) * (3E-3 + 1.75E-2) = 5.19E-6$$

The contribution to core melt due to tornado winds and missiles is $2.53E-4$ without recovery and $5.19E-6$ with recovery. The total contribution to core melt due to straight winds and tornadoes (wind and missiles) is $2.59E-4$ without recovery and $5.31E-6$ with recovery.

7.3 Potential Fixes and Core Melt Probability Reduction

The contribution to core melt probability of the vulnerable equipment/structures given in Table 6-1 is summarized in Table 7-4. The first column is the core melt probability contribution with recovery incorporated due to both straight winds and tornado winds and missiles. The second column is the contribution to core melt probability after the diesel generator exhaust stacks have been fixed (i.e., protected from failure due to winds). For this case the core melt contribution due to extreme winds becomes negligible.

TABLE 7-4
CORE MELT PROBABILITY CONTRIBUTION OF THE
VULNERABLE EQUIPMENT/STRUCTURE FAILURES
BEFORE AND AFTER PROTECTING DIESEL GENERATOR
EXHAUST STACKS FROM FAILING DUE TO WINDS

Equipment/Structure	Prior to Fix	After Fix**
1. Diesel Generator Exhaust Stacks	5.31E-6	ϵ^*
2. Diesel fuel oil storage tank	ϵ	ϵ
3. Condensate Storage Tanks	ϵ	ϵ
4. Borated Water Storage Tanks	ϵ	ϵ

* ϵ = epsilon $< 10^{-8}$

** = Core melt frequency after fixing the diesel generator exhaust stacks.

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APPENDIX H

LIGHTNING ANALYSIS FOR THE
ANO-1 NUCLEAR POWER PLANT

by

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1.0 INTRODUCTION

Each year nuclear power plants report a significant number of lightning-caused turbine trips, plant upsets, and equipment failures. A review of Licensee Event Reports reveals such examples of lightning-caused events as:

- numerous generator trips and reactor trip
- loss of offsite power and 120 V AC vital buses
- loss of a diesel generator power transformer
- blown motor control center fuses
- unavailability of automatic initiation for HPCI and RCIC
- incapacitated annunciators and transformers

Concern resulting from the frequency and consequences of lightning events has led this program to an examination of the probability of lightning leading to a core melt situation. Although a complete analysis of electrical pathways for lightning access to decay heat removal systems is beyond the scope of this program, a bounding calculation of the probability of such an occurrence is attempted and the results can be compared to the plant vulnerability to other special emergencies. The evaluation of plant vulnerability to lightning strikes and subsequent determination of potential contribution to core melt probability proceeds along the following steps:

- a) Calculation of the lightning frequency at the site.
- b) Calculation of the percentage of damaging strikes.
- c) Evaluation of plant lightning protection.
- d) Evaluation of critical components.
- e) Evaluation of core melt probability.

1.1 Calculation of Lightning Frequency

Conservative estimates indicate that some 100,000 thunderstorms occur within the 48 contiguous states during any one year. It has been observed that flashing rates vary greatly from region to region and also from thunderstorm to thunderstorm. It has also been found that the ratio of cloud-to-cloud and cloud-to-ground flashes vary greatly. However, based on general data, a number of empirical formulae have been developed for interim use to relate cloud-to-ground flash

density to thunderstorm-days. For the United States, these relationships are as follows:

Location	Ground Flash Density ($\text{km}^{-2} \text{ yr}^{-1}$)	Reference ¹
Northern U.S.	0.11T	Horn and Ramsey (1951)
Southern U.S.	0.17T	Horn and Ramsey (1951)
U.S.	0.1T	Anderson et al (1968)
U.S.	0.15T	Brown & Whitehead (1969)

T is the number of thunderstorm-days for the locality of interest.

It should be noted that no one relationship can be singled out as the most reliable; however, as a group they provide a general guide to the correspondence between thunderstorm-days and total ground flash density.

A Nuclear Safety Analysis Center study¹ of lightning problems and protection at nuclear power plants has stated that "Relationships developed between lightning activity and latitude are consistent with the observed low incidence of thunderstorms at the poles and the high incidence of thunderstorms at the equatorial belt. However, it has also been observed that the proportion of cloud-to-cloud flashes increases as latitude decreases. It has also been observed that there is a much higher number of flashes in coastal regions than in non-coastal regions. Each of these factors must be considered in an empirical relationship to describe the density of ground flashes per unit time and area for a particular plant site. Pierce, et al, in 1962 proposed the following relationship for cloud-to-ground flash density in terms of latitude and thunderstorm-days per year, T:

$$N_g = (0.1 + 0.35 \sin \text{latitude}) (0.40 \pm 0.20) T \text{ km}^{-2} \text{ yr}^{-1} \quad \text{Eqn 1}$$

This equation was used to evaluate the relative probability of ground strikes at the various locations investigated."

The U.S. Meteorological Service has charted the number of thunderstorm-days per year within the United States as seen in Figure H-1. For this program, then, we will estimate the number of thunderstorm-days per year for the plant region of concern using Figure H-1. This data will then be applied to Equation 1 to obtain the number of cloud-to-ground flashes per $\text{km}^2 \text{ yr}$. This number will be compared for accuracy to similar computations using the relationships presented earlier in this section.

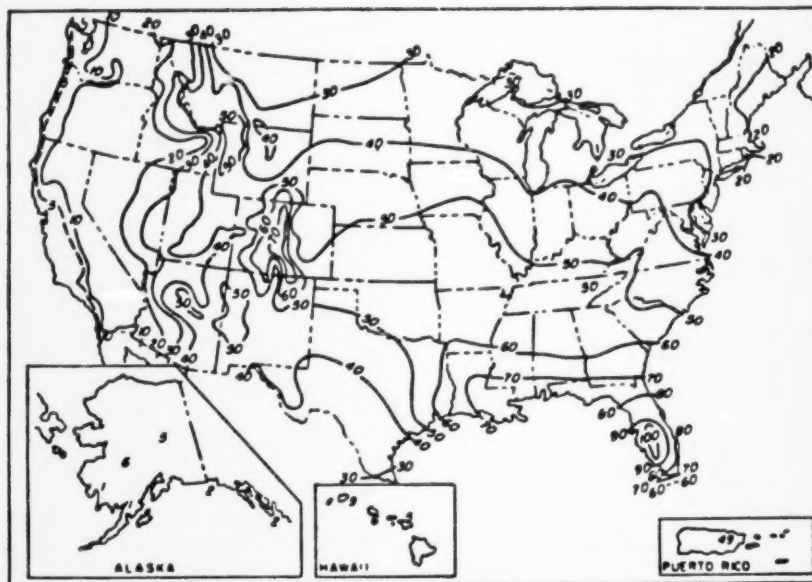


Figure H-1. Map showing the mean annual number of days with thunderstorms. The highest frequency is encountered in south-central Florida. Since 1894, the recording of thunderstorms has been defined as the local calendar day during which thunder was heard. A day with thunderstorms is so recorded regardless of the number occurring on that day. The occurrence of lightning without thunder is not recorded as a thunderstorm. (Data supplied by Environmental Science Service Administration, U.S. Department of Commerce.)

1.2 Calculation of Damaging Strikes at the Site

NRC recommendations suggest that protection be provided for electrical systems against a lightning strike with a current of 200 kilo-ampere (kA). Available information suggests that one out of every thousand lightning flashes has a current in excess of 200 kA. A recent study by Meier² on the frequency of peak lightning current intensities for cloud-to-ground strikes in Florida observed that 1% or less of the flashes observed had currents greater than 100 kA while the largest peak current detected was 418 kA. Figure H-2 gives a cumulative distribution of all strikes as determined by Poplansky in 1972.¹

Since it is not the purpose of this program to evaluate the effectiveness of existing special emergency protection systems, it will be assumed that a plant with existing protection has met the letter of the NRC recommendations and is not vulnerable to strikes of less than 200 kA. (It should be noted however that the NSAC-41 survey of four nuclear power plants revealed a large number of neglected lightning protection systems. Bent air terminals, insufficient grounding, and corroded conductors are just examples of deficiencies they found in existing protection systems.) This is consistent with the analysis of other special emergencies in which fire sprinklers are assumed to function, seismic snubbers are given credit for successful operation, and watertight doors are assumed to hold tight. The scope of A-45 does not include the exhaustive testing or reanalysis of existing special emergency protective systems.

Therefore, plants with existing lightning protection will be assumed to be protected against lightning strikes of less current than 200 kA. Furthermore, it will be estimated that 1% of all lightning strikes exceed 200 kA regardless of the plant's geographical location.

Using the equations in the previous section to determine the frequency of cloud-to-ground strikes per km² yr, and multiplying this number by .01 will yield the number of significant strikes/km² yr for a protected site.

1.3 Evaluation of Plant Protection

As previously stated, it is beyond the scope of this program to test or reanalyze lightning protection at nuclear power plants. Rather, this evaluation of plant protection will be limited to a literature review and a plant visit.

The literature review extends only to information available in the public domain. Studies performed by the plant are combined with any details provided in the Final Safety Analysis Report to determine if lightning protection is available at the site. This review is confirmed during a site visit where the building air terminals, ring conductors, and earth shield lines are

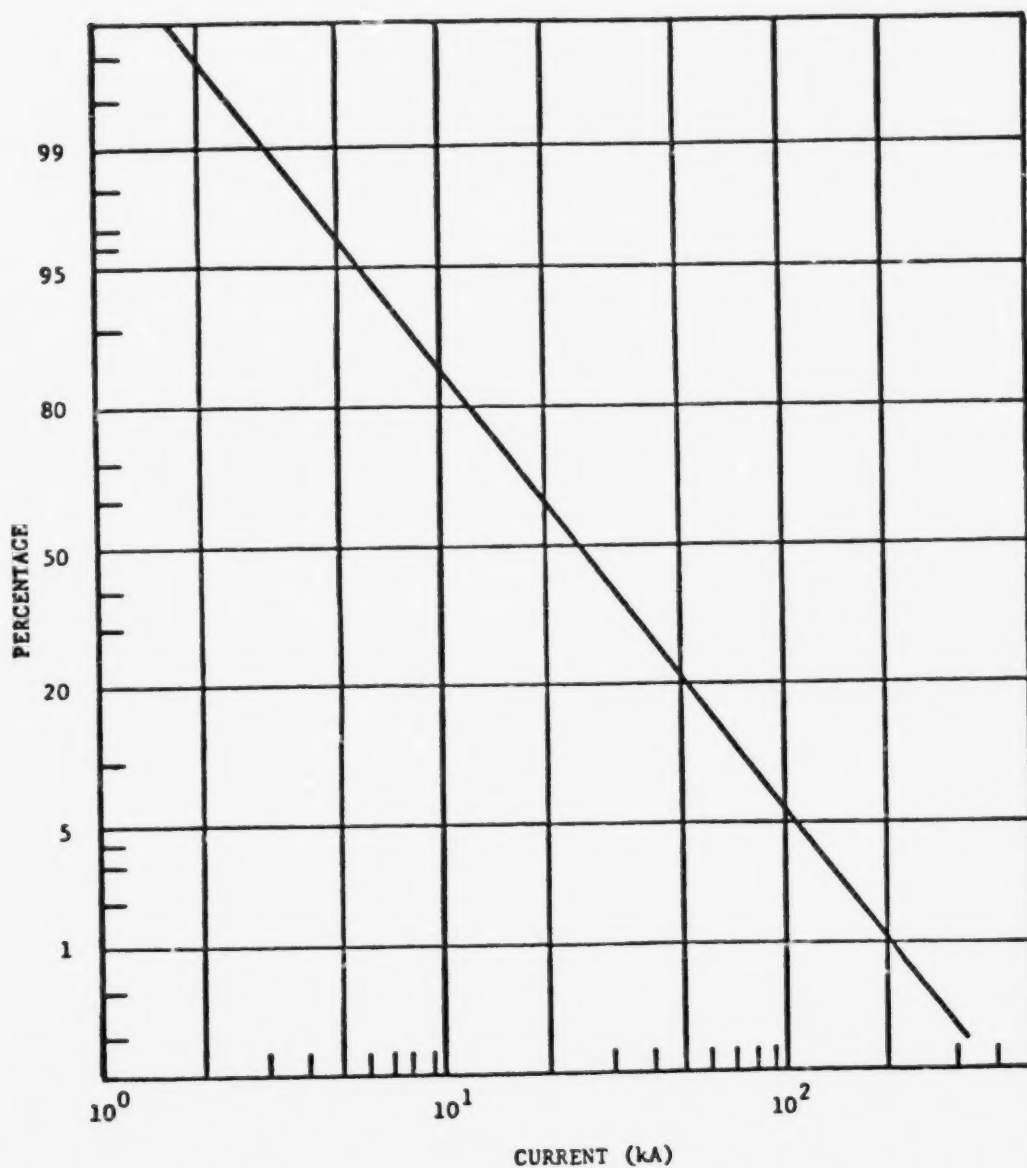


Figure H-2. Cumulative frequency distribution of lightning current amplitudes.

visually confirmed. Additional questions are asked concerning the ground grid system. If these measures are available, credit is given for lightning protection. It should be noted that lightning has been known to enter buildings via ventilation ducting, roof drains, and roof antennas to name just a few paths. These pathways are too numerous and subtle, however, for this program to evaluate and quantify in depth.

1.4 Evaluation of Critical Components

A review of relevant Licensee Event Reports attributing plant upsets to lightning strikes is provided in this Appendix for detailed review of historical evidence of lightning damage. Examples of system effects include:

- loss of offsite power
- loss of 120 V vital AC buses
- false main steam line isolation and safety injection
- incapacitated annunciators and transformers
- loss of DC power transformers
- failure of battery chargers
- failure of motor control center fuses
- loss of circulating water pump
- failure of charging pumps to start
- unavailability of automatic initiation for HPCI and RCIC
- loss of B train 4160 V emergency bus.

These examples reveal that no system or component can be readily exempted from consideration for lightning vulnerability. The approach, then, is to use the transient event trees from the internal event analysis to determine the success criteria of decay heat removal systems needed following an initiating transient. It is easily conceivable that a loss of offsite power condition will exist due to a storm at the time the lightning strikes the plant. A reactor trip has been the result of many lightning strikes due to such conditions as perturbations in sensors, voltage mismatches, or loss of feedwater pumps. Therefore, the transient event trees are a reasonable source to identify the critical front line and support system combinations necessary for successful decay heat removal.

Although all plants require redundant independent trains of decay heat removal which cannot all be damaged by a single

lightning strike, the LERs prove that multiple strikes can hit a plant within a few minutes. Additionally, one train could conceivably be damaged by lightning while the redundant trains could suffer random failure events. Therefore both multiple lightning strikes and single strikes in conjunction with other random failures must be analyzed for their contributions to core melt.

1.5 Evaluation of Core Melt Probability

Once the frequency of lightning strikes with a current greater than the lightning protection design basis has been estimated, and the success criteria of systems needed to respond to a transient have been identified, it is necessary to postulate the probability of a strike damaging one of these essential systems. Given the numerous available lightning targets at a site, as well as the historical evidence of the types of systems that have been hit, it would be considered extremely conservative to assume that one strike out of ten, powerful enough to bypass the lightning protection, will hit a critical decay heat removal system or support system. An example of this would be a pressurized water reactor which would require only auxiliary feedwater or feed and bleed to prevent a core melt. The probability of lightning striking either of these systems, their electric power buses, or motor control centers is small when one considers all possible targets. If a single component, electrical bus, battery, etc., is the necessary target, the probability of a damaging strike, given all of the available targets, is assumed to be one in one hundred (0.01). However, if a single lightning strike could not reasonably fail both systems or trains of electric power, then either a second lightning strike or a random failure would be required to fail the redundant system. As more redundant independent trains are available for decay heat removal, more lightning strikes or random failures would be required for a core melt scenario.

It is probable that plant personnel would effect recovery actions to repair the damage of the lightning strike before core damage could occur. In any case, manual operation of various systems is probable at most plants if power to valves, instrumentation, motor control centers, etc., were to fail. The probability of nonrecovery of damaged systems in time to prevent a core melt is exceedingly remote but will be assumed to be 0.1 as an upper bound for the problem.

The probability of core melt due to lightning would then be calculated by the following equation:

$$\text{Core melt frequency} = T \left[\left(\frac{N_g}{T} \right) (A) (P_{kA}) (P_{hit}) \right]^n (NR) +$$

$$(R_1) T \left[\left(\frac{N_g}{T} \right) (A) (P_{kA}) (P_{hit}) \right]^{n-1} (NR) + \dots +$$

$$(R_1)(R_2) \dots (R_{n-2})(R_{n-1}) T \left[\left(\frac{N_g}{T} \right) (A) (P_{kA}) (P_{hit}) \right] (NR)$$

where T = Number of thunderstorms-days per year for the region

N_g = Number of lightning strikes per year per km^2

A = Area of the plant site in km^2

P_{kA} = Probability that the lightning current is greater than the design basis.

P_{hit} = Probability that lightning hits a critical component

NR = Probability of nonrecovery (0.1)

n = Number of redundant, independent trains available available to prevent core melt

R_n = Probability of random failure of the n_{th} train

The actual plant specific calculations are straightforward and, therefore, given in Section 3.6 of this report.

1.6 Licensee Event Reports (LER) Related to Lightning

Battery Charger Fails During an Electrical Storm at Vermont Yankee

Description. Date of event - 6/78. Power level - 99%. During a severe electrical storm, control room personnel reported a "Battery Charger CA-1 AC Failure" alarm. Investigation by operating personnel determined that the AC input breaker to the CA-1 battery charger had tripped open. About 15 minutes after the trip condition had occurred, the AC input breaker was reclosed with no further incidents reported. This event was caused by an apparent spurious trip of the AC input breaker on AC&D Company Model No. ARC 130H/K150 battery charger. Cause: spurious trip of input breaker. LER/Docket No. LER 78-002 01T-1 Docket 50-293.

Air Sampler Found Inoperable at Vermont Yankee

Description. Date of event - 6/78. Power level - 99%. During the weekly collection of air particulate samples for the period 6/18-6/25 it was discovered that 3 air samplers had been inoperable for the entire week due to a loose fuse. The 3 air sample pumps stopped operating at about the same time during a severe thunderstorm. Therefore, the blown fuses are attributed to the storm. It is believed that the loose fuse was caused by pump vibration. The blown fuses were replaced and the loose fuse tightened. Cause: fuses blown during thunderstorm. LER/Docket No. 50271.

Stack Gas monitoring System Fails at Vermont Yankee

Description. Date of event - 6/78. Power level - 100%. Due to an impending electrical storm, the stack gas II monitoring system was taken out of service in accordance with the station operating procedures. Subsequently, the operating stack gas I monitor failed due to the lightning strike. The high voltage power supply was repaired and the log count rate board replaced. The stack gas monitors are Model 842-1, manufactured by Victoreen Instrument Division. A plant alternation is being installed that will prevent and/or mitigate the consequences of a lightning strike. Cause: lightning strike. LER/Docket No. 50271.

Power to a Motor Control Center Lost at Vermont Yankee

Description. Date of event - 6/78. Power level - 100%. During a severe electrical storm, control room personnel reported a loss of voltage to motor control center 89B. This MCC powers the "B" LPCI injection valves and the "B" recirculation system discharge and discharge bypass valves. The loss of voltage to MCC 89B was caused by the tripping of an Exide power system 250 KVA uninterruptible power supply. Troubleshooting of the tripped unit did not reveal equipment malfunctions or the cause of the unit trip. Following the energization of the UPS-1B unit, satisfactory valve operability surveillance testing of the "B" LPCI subsystem valves was performed and the unit placed in service. Cause: uninterruptible power supply tripped. LER/Docket No. 50-271.

Power to Reserve Transformer Lost at Monticello

Description. Date of event - 6/78. Power level - 60%. During power operations, a lightning storm resulted in plant substation lockouts deenergizing auxiliary reserve transformer 1AR. The relay failed due to short circuit failure of a zener diode in the trip pulse isolator output. This failure is believed to have occurred due to a voltage spike on the 125 VDC supply produced by lightning effects. Zener was replaced with

type 1.5 KE200 having a higher surge rating. Cause: lightning effects. LER/Docket No. LER 78-012 03L-0 Docket 50-263.

All Meteorological Indications Fail at Crystal River 3

Description. Date of event - 7/78. Power level - 0%. All digital and recorded meteorological indications failed contrary to tech spec. Printed circuit cards to the translator power supply and the dew cell were renewed. The cause of this event can be directly attributed to lightning. Cause: lightning. LER Docket No. LER 78-036 03L-0 Docket 50-302.

Loss of All Off-Site Power Scrams the Reactor at Pilgrim 1

Description. Date of event - 8/78. Power level - 100%. A reactor scram occurred as a result of loss of all off-site power. The diesel generators functioned as required and assumed emergency loads, the HPCI and RCIC were manually initiated for water level control and relief/safety valve 203-3C was manually opened once to control pressure and maneuver the reactor to a safe condition. The cause of this event was lightning striking the off-site distribution lines causing a backflash which resulted in high speed relay operation for line-to-ground fault. Power was restored to the startup transformer and station loads returned to normal source after about 1/2 hour. Cause: lightning striking distribution lines. LER/Docket No. LER 78-035 01X-0 Docket 50-293.

Both Roof Vent Flow Recorders Found Reading Downscale at Peach Bottom 3

Description. Date of event - 7/78. Power level - 99%. During power operation the A and B roof vent flow recorders were observed to be reading downscale after an apparent lightning strike. The A monitor recovered without any corrective action within one hour. The cause of the downscale indication was attributed to a lightning strike on the vent stack. The instrumentation was checked and found to be satisfactory except for the indicator drive mechanism on the B flow recorder, which was disengaged. The recorder was repaired, operated satisfactorily and returned to service within ten hours. Cause: lightning strike. LER/Docket No. LER 78-015 03L-0 Docket 50-278.

Lightning Strike Causes Loss of Two Off-Site Power Sources at Prairie Island 1

Description. Date of event - 9/78. Power level - 100%. With both units at full power, a lightning strike on the Spring Creek transmission line caused breaker 6H5 to open. No. 10 bank transformer then tripped on sudden pressure. Two of the three sources of off-site power were now lost and the safeguards buses were now powered from No. 11 cooling tower

transformer through bus ties. The diesel generators started automatically and ran unloaded (no loading was required). The second source of off-site power was restored. No. 10 bank transformer should not have tripped on sudden pressure due to the external fault. A design change in transformer protection relaying is being pursued to prevent recurrence. Cause: design deficiency. LER/Docket No. LER 78-018 01T-0 Docket 50-282.

Lightning Strikes Cause Reactor Trip and Safety Injection at Farley 1

Description. Date of event - 6/78. Power level - 100%. Lightning caused 1A circulating water pump to trip. While reducing power, both feed pumps tripped resulting in reactor trip. Lightning caused the supply breakers to S/U transformer 1B to open resulting in loss to B train buses. Loss of 120 V AC vital buses 1C, 1D (control and protection channels III, IV) resulted in SI and main steamline isolation. All D/Gs started. B train D/Gs picked up required buses. Charging pump 1C and RHR pump 1A did not start. Unit shutdown and bus loss resulted from lightning striking near 1A circ water pump and 230 kV switchboard. SI resulted from loss of 120 V AC vital buses 1C, 1D. The cause of the sequencer failure to pick up charging pump 1C, and the failure of RHR pump 1A to start was not determined. Cause: lightning. LER/Docket No. LER 78-033 01T-0 Docket 50-348.

Main Stack Sampling System Lost at Peach Bottom 2

Description. Date of event - 6/79. Power level - 100%. During an electrical storm, both units at power, lightning struck in the vicinity of the main stack. The result was loss of the main stack sampling system, initiation of HPCI and RCIC on Unit 3, and trip of Unit 3 startup and emergency feed. Shutdown of Unit 3 was begun due to unavailability of automatic initiation capability of HPCI and RCIC. Fuse replaced. Shutdown of both units was initiated due to loss of stack sampling system. Shutdowns were terminated upon discovery and replacement of blown fuses. Lightning-induced electrical transient caused blown fuses on stack sample solenoid valve circuit and also on inverter supplying power to HPCI and RCIC initiating instrumentation. Cause: lightning strike. LER/Docket No. LER 79-031 Docket 50-277.

Reactor Scram from Loss of All Off-Site Power at Pilgrim 1

Description. Date of event - 7/79. Power level - 100%. A reactor scram occurred as a result of loss of all off-site power. The diesel generators functioned as required and assumed emergency loads. Relief/safety valve 203-3D was manually actuated to control pressure and maneuver the reactor to a shutdown condition. The cause of this event was lightning

striking the off-site distribution lines causing a backflash that resulted in a high speed relay operation for line-to-ground fault. Power was restored to the start-up transformer and station loads returned to normal source after approximately 1-1/2 hours. Cause: external cause. LER/Docket No. LER 79-027 Docket 50-293.

All Off-Site Power Lost at Pilgrim 1

Description. Date of event - 8/79. Power level - 100%. A reactor scram occurred as a result of a loss of all off-site power. The emergency diesel generators functioned as required and assumed emergency loads. A relief valve was manually actuated to control pressure and maneuver the reactor to a safe shutdown. The cause of this event was lightning striking the station switchyard causing a high speed relay operation for a line to ground fault. Power was restored to the start-up transformer and station loads returned to normal after approximately 30 minutes. Cause: lightning strike in switchyard. LER/Docket No. LER 79-033 Docket 50-293.

Meteorological Instruments Inoperable at Salem 1

Description. Date of event - 10/79. Power level - 000%. During the refueling outage, morning readings showed meteorological instruments inoperable. The required channels were out of service for seven and one-half hours. No atmospheric radioactive releases were made during this time. This is the second occurrence of this type. An apparent lightning strike near the instrument building caused induced current in a rain gauge sensor which fed back to the 15 VDC power supply, removing power from all sensors. The damaged electronic components were replaced and the possibility of installing surge protection for all channels is being investigated. Cause: power surge from lightning. LER/Docket No. LER 79-066 Docket 50-272.

Wind Direction Channel Fails Upscale at Browns Ferry 1

Description. Date of event - 1/80. Power level - 000%. While Unit 1 was shut down for refueling, wind direction channel at 33 feet elevation as observed on wind direction recorder XR-90-102-2 failed upscale. Wind direction indicators at 150 feet and 300 feet elevations were inoperable. There have been no previous failures of this nature. The series 312D millivolt-to-current converter was repaired and returned to service in five hours. The indicator failed due to lightning damage at the meteorological tower. Cause: lightning damage. LER/Docket No. LER 80-002 Docket 50-259.

All Off-Site Power Lost at Prairie Island 2

Description. Date of event - 7/80. Power level - 100%. With Unit 2 at 100% power and Unit 1 at cold shutdown, a severe electric storm caused operation of several breakers in the substation. The main generator was separated from the grid and the reactor and reactor coolant pumps were tripped. Natural circulation was established for core cooling. At this point two off-site power sources still existed. No. 10 transformer was locked out and with only one off-site source remaining (a degraded one), the diesel generators started and closed in on the safeguards buses. No safety injection was actuated. Not repetitive. The cause was an electrical storm. The CT11 source was restored and the No. 10 transformer was restored. The next day conditions at the substation were normal. Cause: electrical storm. LER/Docket No. LER 80-020 Docket 50-306.

Off-Site Power Lost at Indian Point 2

Description. Date of event - 6/80. Power level - 100%. During normal plant operation, all off-site electrical power was lost to the site. The reactor was shutdown by the operator via manual turbine trip. The generator was subsequently shutdown by a direct trip signal from the Buchanan substation. The reactor was maintained in a safe shutdown condition using natural circulation cooling until restoration of off-site power. A lightning strike resulted in failure of a system transmission tower shield line which in turn fell and faulted electrical feeders W93, W98, 96951 and 96952. The feeders were cleared of the faulted condition and restored to normal service. Cause: lightning. LER/Docket No. LER 80-006 Docket 50-247.

Off-Site Power Lost at Indian Point 3

Description. Date of event - 6/80. Power level - 60%. The unit experienced a loss of outside power on the 138kV feeder. At this point, diesel generators Nos. 32 and 33 were activated to assume the 480V buses. A similar event occurred on July 13, 1977. The cause of the external power loss was determined to be a lightning strike on the ground lead of a transmission tower. The lead became dislodged and fell across the feeder, causing the breaker to open. Repairs were effected and external power was restored. Cause: lightning hit transmission tower. LER/Docket No. LER 80-008 Docket 50-286.

Off-Site Electric Power Line Trips at Browns Ferry 1

Description. Date of event - 8/80. Power level - 73%. Lightning apparently caused Athens 161V line to trip momentarily. After one minute, cooling tower transformer No. 1 tripped causing loss of Unit 3 4kV bus tie board. After about 30 minutes the transformer was reenergized and it tripped again.

Transformer was removed from service for testing and Athens 161kV line reenergized. There were no indications of what tripped the transformer. Cause: lightning strike. LER/Docke No. LER 80-059 Docket 50-259.

EVENTS TAKEN FROM "NUCLEAR POWER EXPERIENCE"

Lightning Strike - Cooling Tower Power Outage, High Blowdown Prairie Island 1 and 2, June 78 - 90% Power

A lightning strike on the plant switchyard caused a power outage to 3 of the 4 cooling towers and a complete loss of power to the blowdown flow control gates. As a result, for a period of approximately 3 hours the circulating water system operated in the helper cycle mode (1 tower operating) with no recirc flow back to the plant.

Meteorological Inverter Switching Mechanism Malfunctioned, Lightning Damaged Power Supply Salem 1 - Mar 78 and Oct 79 - 100% Power and Shutdown

In December 1978, instrument surveillance at the meteorological tower found that the normal power supply to the inverter had failed and the inverter had not shifted to the battery supply. the inverter was manually shifted to the batteries and proper operation of the instruments was confirmed.

Lightning Burned Out Zeners - Transmission Lines Out Kewaunee - Aug 73 (hot functional testing)

During a severe lightning storm that struck in the vicinity of their substation, 3 of 4 transmission lines serving the substation tripped due to burn-out of the Zener elements in the Westinghouse relay circuitry. The elements fully meet proposed IEEE standards. Recommended they remove the Zeners from the circuits.

Lightning Strikes, Loss of Off-Site Power - System Blackout, Condenser Rupture Discs Blown Indian Point 3 - May and July 77 - 91% Power

The 138kV feeder supply outside power to the unit via the station service transformer was lost in May 1977. The loss of outside power was the result of a lightning strike on a feeder in the transmission system. The DGs started and assumed the loads of 480V Buses 5A and 6A. The momentary loss of power to these buses with the resultant loss of NIS Power Range Channel 43 caused a turbine runback to 750 MW. A reactor shutdown was initiated pending evaluation of the incident. After determining the availability of outside power, the shutdown was halted and the unit returned to power. The 138kV feeder was subsequently returned to service.

This occurrence was one of a series of events that led to a blackout of the New York City area. The principal components of the Con Ed system that were damaged consisted of 5 transformers, 10 termination connections, and 15 failed lightning arresters. One of the failed transformers was in the Buchanan Switchyard that provides off-site power for Indian Point 3.

Lightning Caused Loss of Auxiliary Power, Reactor Trip
Cook 1 - Sept 77 - 100% Power

During an extreme thunderstorm, a 345kV transmission breaker faulted and while clearing, lightning hit on a transmission circuit. The protection relaying, to clear the faulted breaker, opened all breakers that could feed to the faulted breaker. This deenergized the transformer from which unit 1 received its normal reserve auxiliary power. Unit 1 was operating at the time, supplying its own auxiliary power, so the loss of the offsite transformer was only the loss of a standby source.

Later, lightning hit and the resultant high-speed opening and reclosure of the associated line breakers on another transmission circuit caused a voltage dip to 64% of normal. The undervoltage protection relays on reactor coolant pump motor buses saw this dip and called for the reactor to trip. The reactor trip cascaded into a turbine trip. By design, the generator trip was delayed 30 seconds after a turbine trip if the initiating trip was a reactor trip. With the generator trip the auxiliary power was automatically transferred from the normal source to the normal reserve source which was deenergized from the earlier incident. Upon loss of voltage, all four 4kV emergency buses shed nonessential loads, the 2 emergency DGs started and accepted their sequential loading within design parameters.

Circulating Water Pump, Start-up Transformer Lost by Lightning
Strikes - Scram, Inverter Fuse Failed, SI
Farley 1 - June 78 - 100% Power

Circulating water pump 1A tripped after lightning hit near the pump. The reactor operator initiated a rapid load reduction to prevent a turbine trip due to the rapid increase in condenser backpressure caused by the reduction in circulating water flow. The resultant transient on SG water levels and FW flow caused a low suction pressure trip of both main FW pumps. The reactor tripped when a low FW flow less than steam flow coincident with low SG water level condition occurred. About 18 minutes later, lightning hitting the 230kV switchyard resulted in the loss of start-up transformer 1B when circuit breakers 820 and 924 opened. This caused a loss of power to B Train 416V emergency buses 1G, 1J, 1L, 600V load center 1E, 1J, 1L, 1S, and 416 buses 1B and 1C. All DGs started, and the B train DGs closed

on 4160 V emergency buses 1G, 1J, 1L, and accepted load. The Tech Specs required that 4160V emergency buses 1G, 1J, 1L, and 600 V load center 1E, 1J, 1L, 1S be operable, energized and aligned to an operable DG. Transformer 1B was restored to service in approximately 10 minutes.

Lightning Blew Diesel Transformer Fuses
Oyster Creek - June 73

During a plant start-up, with plant electrical loads being supplied by off-site power, a lightning storm occurred in the vicinity of the plant causing an apparent voltage surge on the 1B 4160V bus blowing the 3 power transformer fuses for diesel generator No. 2. The result was to render the fuel oil transfer pumps inoperable since they are among the auxiliaries supplied by the transformer. Although the unit would have started and been operable if a loss of off-site power had occurred, its fuel supply would have been limited to the quantity in the day tank.

Lightning Knocked Out Reserve Transformer - Relay Scheme Modified
Monticello 1 - June 76 - 100% Power

A loss of power to a reserve transformer was experienced. The power loss was due to a lockout of the substation transformer. It was suspected that lightning-induced voltages caused arcing across the sudden pressure protection relay contacts. The relay scheme was modified to use a set of normally closed contacts on the sudden pressure relay to short the coil operated by this relay when not needed.

Lightning Strike Damaged Breaker
Peach Bottom 2 - June 75 - 960 MWe

A lightning strike on the 500 kV system line caused a generator lockout with an accompanying turbine trip and reactor scram. Significant damage was sustained on the 215 breaker.

Storm-Related Voltage Transients Caused DEH False Signals
Cooper - Aug 80 - 86% Power

A severe weather storm caused a line fault and it was believed subsequent voltage transient signals were induced into the turbine generator Digital Electrical Hydraulic (DEH) system computer. The reactor scrambled automatically when the MSIVs closed.

The DEH computer circuitry sensed false signals indicating the TG output breaker opening. This signal caused the bypass valves to open. With the bypass valves open and the governor valves not controlling the main steam pressure, a Group I isolation occurred due to low pressure and the reactor scrambled

on MSIV closure. Further evaluation of the electrical circuitry was planned.

Explosion in Off-Gas System - May Have Been Caused by Lightning
Quad Cities 2 - Mar 73

An OCB in the 345 kV switchyard tripped. The cause was not immediately apparent, however, it was believed to be the result of lightning from a storm in the area. At the same time, the control room operator noticed that the air ejector discharge flow recorder was oscillating in an erratic manner. A power reduction was initiated and then stopped when the flow indication stabilized after 1 minute.

A few minutes later, several area radiation monitors in the turbine building began alarming in the control room and the standby gas treatment system started automatically when its 2 mr/hr setpoint was exceeded. Combined with the other indications the shift engineer was relatively certain that the off-gas rupture disc had blown. The shift foreman and several operators entered the turbine building dressed in protective clothing and full face masks. An inspection of the 2B air ejector room confirmed that the rupture disc had in fact blown and the operators began placing the 2A air ejector in service and isolating the 2B ejector.

Although the exact cause of the detonation was not positively identified, it was believed to have been the result of lightning in the vicinity of the station.

Lightning Caused Explosion in off-Gas System
Vermont Yankee - June 73

While at 78% power with the interim Off-gas (IOG) system in service, and during a severe electrical storm in which a direct lightning strike to the top of the plant ventilation stack was observed, a detonation in the off-gas system caused sufficient back pressure to fracture the air ejector rupture disc, thereby degrading a boundary designed to contain radioactive materials as defined in the tech specs. Concurrently, the area gamma radiation monitor on the perimeter fence became inoperable. This instrumentation is required at all times by the tech specs. The cause of the detonation and instrument malfunction was believed to have been the lightning strike.

Lightning and Possible Smoldering Material Caused 2 Off-Gas
System Explosions
Vermont Yankee - Aug & Sept 73

While at 50% power and during a severe electrical storm in which a direct lightning strike to the top of the plant ventilation stack was observed, both plant stack gas monitoring systems and the area gamma radiation monitor on the perimeter fence became

inoperable. Since this instrumentation is required at all times by tech specs, an orderly plant shutdown was initiated.

Lightning Knocked Out Plant, Stack Monitors, Etc.
Vermont Yankee - July 74

At 80% power during a severe electrical storm direct lightning strikes to the top of the plant ventilation stack were observed. Both plant stack monitoring systems became inoperable. About 30 seconds later, another strike resulted in a generator/turbine trip which caused a reactor scram. The perimeter fence monitor, the stack base fan, the process computer, miscellaneous annunciating and indicating circuits, 5 auxiliary equipment transformers, the stack particulate tape transport, the No. 1 fan in the east cooling tower, and the instrumentation from the site meteorological tower were rendered inoperable.

In an effort to prevent the consequence of a direct lightning strike to the plant stack and the problems associated with a plant shutdown without the stack gas monitors in service, they were considering the following:

- Install an additional grounding network at the plant stack that will be connected to the existing plant ground network.
- Revise the applicable plant procedure to provide a means for operating personnel to electrically disarm and disconnect the cables from 1 of 2 stack gas monitors when an electric storm appeared imminent. This would ensure that at least one of the monitors was operable following an electrical storm. This was considered to be an interim measure.
- Install electrical surge suppressors in the circuits of specified control room annunciators.
- In an attempt to remove potential sources of conductance from the stack to the plant buildings, the stack base fan indicating lights be removed from the control room.
- Fabricate a calibration chamber that shields the stack gas detectors from background radiation levels when the detectors are being source calibrated.
- Alter the power source inputs to the monitors in such a manner that one of the monitors may be removed for repair without interrupting the service of the remaining monitors.
- Continue investigation of component failures resulting from severe electrical storms.

The duration of the outage was $\approx$ 3 days.

Lightning Strike Caused Stack Monitoring System Failure
Vermont Yankee - May 77 - 99% Power

Due to an impending electrical storm, the stack gas II monitoring system was taken out of service. A short time later a direct lightning strike to the top of the plant stack was observed.

Both stack gas monitoring systems failed due to the lightning strike. Stack gas I had a failed log CRM board while stack gas II had a blown control power fuse. The stack gas monitors were Victoreen Model 842-1. Stack gas II was returned to service within 1 hour. An evaluation was to be made of the lightning protection for the control power transformers to the stack monitoring systems.

Lightning Knocked Out Environmental Sample Station
Vermont Yankee - Sept 80 - 98% Power

During weekly environmental air sample collections it was discovered that a continuous sample was not being drawn at Station AT2.3.

The main fuse for the station had blown out ≈8.7 hours into the sampling cycle. A severe electrical storm was reported in the area at that time. A new fuse was installed in the sample station and a subsequent functional check of the station was performed. No further damage was noted. If feasible, lightning surge suppressors were to be installed in all 9 environmental sample stations.

Lightning Damaged Meteorological Transformer and Transceiver
Nine Mile Point 1 - Aug 80 - 97% Power

A loss of meteorological data transmission to control room instruments was experienced for a 10 day period.

While experiencing a rain and electrical storm, a loss of power to the tower equipment occurred from a lightning strike to its power transformer. To prevent recurrence, spare parts were made more readily available.

REFERENCES

1. NSAC-41, Lightning Problems and Protection at Nuclear Power Plants: pg B-13, December 1981.
2. Maier, M.W., A. G. Boulanger, and R. I. Sox, An initial assessment of flash density and peak current characteristics of lightning flashes to ground in South Florida, NUREG/CR-1024, Office of Standards Development, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

APPENDIX J

IMPACT ANALYSIS FOR THE
ANO-1 NUCLEAR POWER PLANT

by

United Engineers & Constructors, Inc.
Philadelphia, Pennsylvania

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PLANT IMPACT ASSESSMENT REPORT
ARKANSAS NUCLEAR ONE DECAY HEAT REMOVAL ALTERNATIVES

1.0 SUMMARY

Two alternative approaches for reducing the vulnerability of the Arkansas Nuclear One Power Plant to occurrences which could impair its decay heat removal capability were identified by Sandia National Laboratories. In this study, both of these alternatives were evaluated to determine their impact to the owner-operator in the areas of capital cost, operations and maintenance (O&M) costs, O&M complexity and availability. Since the safety value of each of these approaches was the subject of Sandia work, it was not assessed further here.

In summary, the two alternatives examined are as follows:

Alternative 1 consists of the following:

- o Rewiring and piping of the existing auxiliary feedwater pump to facilitate its functioning as an additional emergency feedwater pump. Included in this modification is providing Class 1E power to the pump and adding a new suction line directly to the condensate storage tank with a motor operated isolation valve which can be operated from the control room.
- o Reduce the vulnerability to a station blackout by adding a turbine driven generator which can be aligned to either 4160V switchgear to power the auxiliary feedwater pump and the battery chargers.
- o Install a third decay heat removal pump which can be powered from and supply either train of existing equipment.

- o Provide redundant Borated Water Storage Tank supply valves to ensure adequate flow to the Safety Injection Systems.
- o Install seismically qualified anchors in the 4160V switchgear, 480V transfer cabinets, 480V MCC's and the control room panels to assure the availability of these items following a seismic event.
- o Install seismic bracing on the borated water storage tank to decrease its vulnerability to a seismic event.
- o Install redundant sensing and deluge valve in the cable spreading room fire protection system.
- o Provide additional supports on the exhaust for the diesel generators to prevent their failure during high wind conditions.

Alternative 2 is a complete "add-on" system with single trains for both decay heat removal and reactor makeup. Both trains are housed in new buildings with dedicated water supplies and emergency electrical power.

The results of the impact assessments are summarized in Table 1.0-1. All costs are given in real dollars, levelized over the remaining economic life of the plant and are for Unit 1 of the Arkansas Nuclear One Plant. Costs shown are "generic" in nature and are based on data (unit material costs, labor rates, labor productivities) consistent with DOE's "Energy Economic Data Base (EEDB)"; however, the EEDB labor productivities were

adjusted to account for known "complicating factors" such as accessibility, interferences, congestion and the special health/safety considerations inherent in backfit type work.

It is evident from the table that Alternative 1 which includes minimal new equipment and changes to existing systems or structures can be achieved at a lower cost and has less impact on plant operations and maintenance. This alternative can be achieved with minor radiological exposure. Alternative 2, which adds new structures and a considerable amount of additional equipment, has a major capital cost impact in the range of four times that of Alternative 1.

TABLE 1.0-1
ALTERNATIVES IMPACT SUMMARY
ARKANSAS NUCLEAR ONE DECAY HEAT ALTERNATIVES

<u>Alternative</u>	<u>Annualized Capital Costs</u> (Inflated \$ x 10 ³)	<u>Operations</u>	<u>Maintenance</u>	<u>Radiation Exposure</u> (Man-Rem)
1	2,915	Minimal-some operator training required but day-to-day operations will not be significantly affected.	Minimal-requires less than 225 mh/yr primarily for isolation valve testing and maintenance.	0
2	10,641	Moderate - requires additional line-up checks and significant operator training	Significant - will cost on the order of half a million dollars a year to maintain the added equipment and structures	20

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The operational and maintenance impact is also much more significant and radiological exposure is significantly higher.

The highest radiological exposure is 20 man-rem for Alternative 2.

Since each of the alternatives involves addition of some active equipment which is required to bring the plant to some prescribed level of safety, it is presumed that the unavailability of this equipment, once installed, will be a limiting condition for operations. Thus, it is expected that the Alternatives will serve to reduce plant availability somewhat. Although that reduction was not quantified, it is not expected to be large.

2.0 INTRODUCTION

The purpose of this report is to provide an assessment of the impact of implementing proposed alternatives for improving the decay heat removal capabilities of the Arkansas Nuclear One Power Plant. These alternatives have been identified by Sandia National Laboratories as possible approaches to reducing the projected core melt frequency for this plant. This impact assessment, together with similar assessments for other plants, will be used by Sandia as part of the basis for formulating recommendations to the Nuclear Regulatory Commission to resolve Unresolved Safety Issue A-45, "Decay Heat Removal."

A total of two promising approaches, or as they are termed in this report, "alternatives," were identified by Sandia on the basis of probabilistic risk assessment studies. The first alternative consists of nine somewhat complex modifications. The second alternative is the addition of a new facility equipped with decay heat removal and reactor makeup systems.

Summary descriptions of each of the modifications proposed for Arkansas Nuclear One are provided below and are described in somewhat more detail in Attachments A through I. These attachments are presented as Modification Design Reports prepared by the architect-engineer supporting the Sandia work. Each modification was given an identification number for control purposes; these numbers are included here for convenience in referencing individual modifications. The modifications considered are as follows:

Modification 801 - Modifications to the Auxiliary Feedwater Pump
(Attachment A)

This modification involves rewiring the existing auxiliary feedwater pump to provide a Class 1E power source and modifying the piping configuration to allow its use as an additional emergency feedwater pump. The purpose of this modification is to reduce the vulnerability of the emergency feedwater system to pump, turbine and battery failures.

Modification 303 - Addition of a Turbine Driven Generator for Emergency Loads
(Attachment B)

This modification will reduce the plant vulnerability to a station blackout by providing an additional power source to the auxiliary feedwater pump and the battery chargers. This will increase the availability of the emergency feedwater system and assure the availability of key instrumentation following a total station blackout.

Modification 804 - Addition of a Third DHR Pump
(Attachment C)

This modification involves the addition of a third 100% capacity DHR pump which can be powered from either Class 1E power system. The pump is installed to supply either DHR train with full capacity thereby improving the reliability of the DHR system.

Modification 805 - Addition of Redundant BWST Supply Valves
(Attachment D)

This modification reduces the plant's vulnerability to a loss of the safety injection systems due to valve failures by adding redundant valves in the BWST supply line. This provides two parallel flow paths from the BWST to the safety injection systems thereby improving their availability.

Modification 106 - Improved Seismic Anchorage of Electrical Equipment
(Attachment E)

This modification will provide additional seismic anchorage for certain safety related electrical components, namely, the 4160V switchgear cabinets, MCC's and the main control room panels. This anchorage will assure the availability of this equipment.

Modification 107 - Condensate Storage and Borated Water Storage Tank Supports
(Attachment F)

This modification reduces plant vulnerability to a seismic event by adding braces to the condensate storage and borated water storage tanks. This modification will increase the strength of the tanks sufficiently to ensure their availability following a seismic event.

Modification 811 - Cable Spreading Room Fire Protection System
(Attachment G)

This modification reduces plant vulnerability to a fire in the Cable Spreading room by installing a redundant detection system which activates a redundant deluge valve in the main header. The probability of a fire in the Cable Spreading room will therefore be reduced as will the probability of a common mode failure resulting in loss of multiple decay heat removal systems.

Modification 112 - Diesel Generator Exhaust Supports
(Attachment H)

This modification reduces the plant vulnerability to high winds by supporting the diesel generator exhaust lines to prevent buckling and the resulting impairment of diesel operation.

Modification 812 - Alternative Decay Heat Removal Add-on System
(Reference 1, Appendix I)

The purpose of this modification is to improve the reliability of the decay heat removal function by adding additional, protected redundancy. It involves the addition of a new, complete, and dedicated Decay Heat Removal System to assure core cooling and reactor coolant makeup following a loss-of-coolant accident or external events such as fire, flood, tornado and earthquake. One of its subsystems will supply emergency feedwater to two of the plant's steam generators. Decay heat would be removed by releasing steam through new steam generator atmospheric steam dump valves. The other of its subsystems provides makeup to the Reactor Coolant System in order to maintain inventory. Both of these subsystems are powered from a dedicated diesel and fed from a dedicated water supply.

This "Add-on" Decay Heat Removal System (ADHR) is housed in two new seismic Category I buildings designed to resist tornado missiles. The ADHR building contains the following components: Alternate Decay Heat Removal Pump, Alternate Service Air Compressor, Alternate Diesel Generator and Associated Tanks, Alternate Makeup Pump, heating and ventilating units, valves, instruments and controls. The other new building contains the Borated Water Storage Tank, Alternate Condensate Storage Tank, Tank Circulation Pumps, valves, instruments and controls.

The Alternate Decay Heat Removal Pump takes suction from the Alternate Condensate Storage Tank and discharges into the existing emergency feedwater piping outside containment. The Alternate Makeup Pump takes suction from the new Borated Water Storage Tank and discharges to the safety injection header outside Reactor Containment and then to the Reactor Vessel.

The only way to characterize the three alternatives proposed by Sandia is in terms of the individual modifications which are their constituents. The following matrix provides that characterization:

<u>UE&C Mod No.</u>	<u>Sandia Mod Designator</u>		<u>Alternative</u>	
			<u>1</u>	<u>2</u>
801	Internal	1	X	
303		5	X	
804		6	X	
805		8	X	
106	Seismic	1	X	
107		2	X	
811	Fire	1	X	
112	Wind	1	X	
812	Add-on	1		X

Section 3.0 of this report discusses the methodology and approach which was used to assess the impact that each of the proposed alternatives would have on operations, maintenance, in-service inspection, and radiation exposure and to develop the cost and schedule required for implementation. Section 4.0 presents a discussion of the impact in each of these areas. Section 5.0 provides any general conclusions which can be drawn from this work.

All impact assessments in this report are based on the Arkansas Nuclear One Power Plant.

3.0 METHODOLOGY AND APPROACH

3.1 Design Characterization

The objectives of the impact assessment work were to:

- o Determine the effect that each of the proposed alternatives would have on subsequent operations, maintenance and in-service inspection.
- o Determine all of the costs involved in implementing each alternative. These include not only base capital and replacement power costs needed to install each of the related modifications but also any significant long-term operations, maintenance or in-service inspection costs.
- o Determine both the short-term and long-term radiation exposure which will result from implementing each alternative.

To achieve these objectives it was necessary to first characterize each modification to the extent that such determinations could be made reliably and areas of uncertainty could be identified without a detailed design effort.

This characterization was accomplished in several steps:

- o Initial Characterization - Following Sandia's identification of the modifications to be considered, draft Modification Design Reports were prepared. Each report identifies functions and design requirements, describes the proposed design, establishes component design bases, identifies supporting (interface) services required from other systems and structures, and identifies the scope of work (Bill of Materials, rip-out required, etc.) and major construction activities. This first effort was based on available plant documentation which was found to vary greatly in scope, depth and quality with the age and type of the plant and the plant designer.
- o Site Visit - Based on the initial characterization, a site inspection plan was prepared for each modification. This plan identified the specific areas which were to be investigated on the site visit in order to confirm the suitability of the design of the proposed modification and to collect cost data and the physical data (congestion, access, radiation level, etc.) needed to permit realistic cost and schedule estimates. Site visits typically involved 2-3 days of intensive effort by site review teams consisting of two engineering supervisors (one from each of the two engineering disciplines having the most impact on the design) and one cost estimator each experienced in plant modification work. Data collection involved discussions with plant engineering and operations personnel as well as walkdowns of the systems and areas involved in each modification. In-plant data typically was collected in the form of tape-recorded narrative descriptions and complimentary instant-developing photographs.

- o Final Characterization - Data collected on the site visit was used to finalize the Modification Design Reports which appear as appendices to this report. In some cases the initial design approach was simply confirmed; in others, changes were made to correct misconceptions resulting from insufficient initial data or to incorporate worthwhile suggestions made by the plant staff.

This process was changed somewhat for Modification 812 which involved the Add-on Decay Heat Removal System. A previous study done for Sandia (Reference 1) had characterized this system and related structures in some detail. For purposes of this study the Reference 1 design was accepted and used, as is, except that the airplane crash protection features were removed and power supplied from off-site was added. It was felt that, since the remainder of the plant did not include airplane protection features, the added cost was not warranted. No further reviews were done to assess the acceptability of the Reference 1 design.

3.2 Operations and Maintenance Assessments

Based on the designs, as they are described in the final Design Modification Reports, each of the alternatives was evaluated for its impact on operations and maintenance (including in-service inspection and testing). Since detailed designs were not pursued, full availability models were not available, and funding was limited, these assessments tended to be qualitative rather than quantitative. Thus, they are based on the best judgements of the participating supervising engineers and managers.

3.3 Total Evaluated Costs

3.3.1 General Approach

In developing the cost impact for each modification, the general methodology specified in Reference 2 was used. Specifically, the simplified model approach described in Section 2.4.1 was selected because it provided reasonable accuracy and was more practical in light of the large number of modifications to be considered in the course of the study. The costs incurred by government agencies to specify and assure the implementation of such modifications were not considered.

3.3.2 Economic Ground Rules and Assumptions

To develop total evaluated costs for each alternative, it was necessary to adopt certain economic ground rules and assumptions. Since it was intended to provide these costs on both a local and generic basis, separate data for both cases has been provided in Table 3.3-1. For a more detailed discussion of the rationale and basis for producing both generic and local estimates, see Section 3.3.3.2 below.

Inflation rate, discount rate, fixed charge rate and O&M levelization factors were estimated using methods and economic data from References 3 and 4.

Replacement power unit costs were estimated using unit costs from Reference 5 which are given by season of the year. Since it is not known when during the year the modifications would be accomplished, these seasonal values were averaged over the year. Also, since Reference 5 costs are given in January 1984 dollars, they were escalated at 3% to be consistent with the capital cost base.

It should be noted that Reference 5 unit costs are calculated on the assumption that the unit would have been operating at its average annual capacity factor during the period of evaluation. Thus, a day's extension of a shutdown results in replacement power charges for that day equal to the charges for one full power day times the plant's capacity factor. The length of an average refueling shutdown was obtained from reactor suppliers.

Base O&M labor rates were obtained from the authors of Reference 6. The labor rates shown in Table 3.3-1 are these base rates increased to include the employer's costs for social security and workmen's compensation. The O&M labor/materials ratio suggested in Reference 3 was used along with total labor to estimate total O&M costs.

TABLE 3.3-1
ECONOMIC GROUND RULES AND ASSUMPTIONS

<u>PARAMETER</u>	<u>GENERIC</u>	<u>LOCAL</u>
1. Plant Location	Eastern Pennsylvania	Northeast Arkansas
2. Remaining Economic Life (Yrs)	19 yrs	22 yrs
3. Inflation Rate (%)	6	6
4. Discount Rate (%)	10.5	10
5. Fixed Charge Rate (%)	17.0	16.0
6. O&M Levelization Factor	1.47	1.69
7. Replacement Power Costs (\$/MW(e) - Day)	550 ⁽¹⁾	423
8. Average Refueling Shutdown Length (Days)		
PWR	60	60
BWR	70	--
9. O&M Labor Rate (\$/Hr) ⁽²⁾	22.80	16.02
10. O&M Labor/Materials Ratio	40/60	40/60
11. Owner's Costs (%)	10	10
12. Contingency (%)	25	25

Note: (1) MAAC Power Pool Average

(2) Includes employer's social security and
workmen's compensation contributions

Owner's costs are an allowance to account items such as the following which can be directly related to the owners support activities:

- o Retraining of Operators
- o Engineering support activities such as review and approval of contractor engineering and design work, preparation of procedure and manual changes, interaction with Regulatory agencies, and preparation of as-built drawings.
- o Quality Assurance/Control
- o Health Physics
- o Owner Management

The relationship between owner's costs and capital costs is highly utility and task dependent. Utilities differ on how much they feel they need to be involved in supervising and managing modification tasks, whether they will do any engineering, design, purchasing or construction management themselves and how they choose to account for their activities on a task basis. For this study, it was assumed that an outside agency(ies) would be contracted to do the engineering, design, purchasing and construction management and related costs were included in indirect capital costs. Therefore, owner's costs were assumed to be 10% of capital costs to reflect this approach and a utility with average involvement in following contractor work. This allocation for owner's costs is thought to be representative based on Reference 7 which, in turn, is based on an informal survey of Gas-Cooled Reactor Associates member utilities.

3.3.3. Capital Costs

3.3.3.1 Cost Accounts

Capital cost estimates were prepared for each modification and are presented in Section 4.0. In these estimates the DIRECT COST accounts include the following:

- o All equipment and materials installed in each modification. Equipment and material quantities are as identified in the Modification Design Reports.
- o All labor costs which can be directly accounted for (i.e., not prorated) on a task basis. These include costs for ripout, installation and supporting services such as fire watch, scaffolding erection and dismantling and cleanup. Labor rates are based on direct hire and include social security and workmen's compensation taxes. Where multi-shift work was deemed prudent to keep reasonable crew sizes and avoid extending the shutdown, shift work premiums were added.

The following items are included in the INDIRECT COST accounts:

- o Temporary Construction Facilities
- o Construction Tools and Equipment
- o Local Taxes
- o Engineering, Quality Assurance and Construction Management by Design Organization(s)
- o Field Office, Job Supervision, and QA/QC by the Maintenance Contractor or Constructor
- o Test and Start-up Engineering

3.3.3.2 Types of Estimates

Two types of cost estimates were prepared for each alternative. The first, termed a "Quasi-Generic Alternative Estimate," is designed to show the effect that technical/

configuration differences would have on costs. In this type of estimate, the costs are all derived from a common cost base using an approach intended to minimize the uncertainties associated with cost estimating techniques. Prepared in this way, these estimates should reflect only the physical differences among different alternatives on a given plant and between the same alternative on different plants. The other type of estimate, termed a "Local Alternative Estimate" is intended to reflect costs which the owner might estimate using a cost data base and techniques more in line with their own local experience. Each of these approaches is discussed in more detail below.

- o Quasi-Generic Alternative Estimate - In this type of estimate the component and material specifications and quantities reflect the specific configuration of the alternative and plant for which it has been prepared; however, the cost base from which it is derived does not necessarily reflect the plant owner's experience. It was prepared to allow cost comparisons which reflect only technical differences, not differences due to the owner's cost experience or estimating techniques. For that reason, all estimates of this type use cost data consistent with the same generic data base. The generic data base used is the "Energy Economic Data Base (EEDB)" maintained by United Engineers and Constructors for the

Department of Energy (Reference 8). The EEDB was chosen as the starting point for these estimates because of its long-term credibility and its known relationship to new plant construction experience in the United States.

In estimating installation direct costs, EEDB material unit costs and labor rates were applied directly; component costs were obtained from vendor quotes, the EEDB or the architect-engineer's in-house data base. However, labor productivities from the EEDB, being representative of new construction work, had to be adjusted to accurately reflect "backfit" work on an operating plant. To facilitate the process of making the necessary adjustments to EEDB labor productivities in a consistent and reliable way, a matrix of labor productivity multipliers was prepared. This matrix, which is based on the architect-engineer's in-house cost data base for backfit work, permitted the estimator to select from a range of multipliers depending on the "complicating factors" for that work. These "complicating factors" include things such as congestion, accessibility, interferences, and health considerations such as radiation level, surface contamination levels and temperatures in the work area.

- o Local Alternative Estimate - This type of alternative estimate differs from the Quasi-Generic estimate in that the generic data is replaced by data which better reflects the local experience of the owner in the areas of labor rates, productivities and other costs.

For Modification 812 involving the Add-on Decay Heat Removal System, the cost estimate created for Reference 1 was used as a starting point. No attempt was made to verify material takeoffs since detailed, scaled drawings were not available. This estimate required a number of changes to make it plant specific, convert it to the same cost base as the other modification estimates and to eliminate the cost of aircraft crash protection features. The Reference 1 estimate, having been created some time ago, needed to be brought up to a January 1985 cost base and have labor productivities adjusted to reflect current industry experience for backfit work. Also, the estimate was adjusted to account for plant size and physical configuration where they differed from those of the plant used as the basis for Reference 1.

For all, except Modification 812 involving the Add-on Decay Heat Removal System, it was assumed that the modification work packages would be relatively small and on a short schedule and thus would be paid for from retained

earnings. The Add-on Decay Heat Removal System in Modification 812, however, being a much larger single capital investment package with a much longer schedule would most likely incur both escalation and interest charges. Thus, both escalation and interest during construction charges were added to Alternative 2 which consists of Modification 812.

Base capital costs were annualized by multiplying by the appropriate fixed charge rates.

3.3.4 Replacement Power Costs

Total replacement power costs were estimated using estimated replacement power unit costs shown in Table 3.3-1 and the projected additional time, beyond that required for an average refueling shutdown, to complete the modification for each alternative.

The need to extend a refueling shutdown was determined on a case basis. Rough durations for work that needs to be done with the plant shutdown (see Section 6.0 of the Modification Design Reports) were first developed based on manhours in the first-cut capital cost estimates and reasonable crew sizes. Added costs for multiple-shift work needed to ensure all work is completed within one or consecutive (where practical) shutdowns were then estimated. Then, considering the cost of replacement power, a decision on multiple-shift

work was made. Where the shift work differential was small (on the order of a few days of replacement power), shift work would be chosen on the assumption that the plant would opt for shift work early in the job anyway in order to not risk extending the shutdown. For longer periods, the decision was made based on the cheaper option.

Replacement power costs, being one-time charges incurred during the outage, were included in total evaluated costs as capital expenditures and annualized using the appropriate fixed charge rate.

3.3.5 Operations and Maintenance Costs

The added operations and maintenance manhours were estimated for each modification and then accumulated for each alternative. These manhours were converted to levelized annual operations and maintenance costs using labor rates, levelization factors and the operations and maintenance labor/materials ratio shown in Table 3.1-1.

Separate O&M costs were calculated for the generic and local cases.

3.4 Radiation Exposure

General area radiation level data was obtained for those areas expected to contribute significantly to accumulated dose. This data, gathered either during the site visit or in subsequent telephone

calls, was used together with cost estimate manhour data to obtain total man-rem exposure for each modification. Based on a review of the proposed work, judgements were made by the engineers who made the site visit regarding the amount of work that could be pre-fabricated and what would have to be done in place. The latter, together with the related general area radiation levels, was used directly to obtain the exposure for each modification. The total exposure for an alternative was then obtained by accumulating the doses for each related modification.

Lacking detailed information on radiation levels and not having a detailed work plan, it was not practical to assess whether any temporary shielding would be appropriate. Thus, none was included in the cost estimate. However, there is sufficient contingency to accommodate any shielding should it be required.

4.0 IMPACT OF PROPOSED ALTERNATIVES

4.1 Alternative 1

4.1.1 Operations and Maintenance

The discussion below deals only with impact on operations and maintenance activities which follow full implementation of the modification. Impacts on plant schedules and costs incurred in installing the modification are treated in Section 4.1.2. For purposes of this discussion "Operations and Maintenance" also includes in-service inspection (ISI) and periodic testing.

Alternative 1 consists of all the in plant Modifications (801, 303, 804, 805, 106, 107, 811 and 112).

Modification 801 involves modifications to the existing auxiliary feedwater pump power supply and piping so it can function in parallel with the emergency feedwater pumps. The auxiliary feedwater pump will be isolated from the CST by a motor operated valve which will require in-service inspection and periodic testing. In addition, since the pump is now a backup to the emergency feedwater pumps, it will also require periodic testing.

Modification 303 adds a turbine driven generator to power either 4160V switchgear via transfer switches. The valves, turbine, generator and transfer switches will require in-service inspection and periodic inspection.

Modification 804 adds a third DHR pump which will require inservice inspection and periodic testing similar to the existing pumps.

Modification 805 adds redundant BWST supply valves on the safety injection supply header. These valves will require in-service inspection and period testing the same as the existing valves.

Modification 811 adds a redundant sensing system and deluge valve on the cable spreading room fire protection system. The detection and actuation systems will require periodic testing at the local panel.

These modifications add equipment designed to raise the level of safety. Since their loss is expected to be a limiting condition of operation, the availability of the plant should be reduced slightly.

The other modifications in Alternate 1 (106, 107 and 112) are structural and will require a minimum of inservice inspection. It is estimated that Modifications 801, 303, 804, 805 and 811 will require a total of approximately 2000 manhours per year for in-service inspection, testing and maintenance.

4.1.2 Total Evaluated Costs

Total evaluated annual costs estimated to result from this alternative are as shown in Tables 4.1-1 and 4.1-3 for the generic and local cases respectively. Associated capital costs showing a breakdown of direct costs by modification and total indirect costs for the entire alternative are shown in Tables 4.1-2 and 4.1-4. Owner's costs and contingency were calculated in accordance with the ground rules discussed in Section 3.3.2. These modifications can be completed in parallel while the plant is on-line or during a single shutdown. There are no replacement power costs. Base capital costs, contingency, owner's costs and replacement power costs were capitalized and fixed charged at rates of 17 and 16.0 percent for the generic and local cases, respectively. O&M costs were calculated using estimated hours identified in Section 4.1.1 and the method described in Section 3.3.5.

4.1.3 Radiation Exposure

Installation of the modifications in this alternative will result in a minimum radiation exposure for plant personnel.

TABLE 4.1-1
TOTAL EVALUATED COSTS FOR
ALTERNATIVE 1 - GENERIC BASIS
(Dollars in Thousands)

1. Base Capital Costs (1/85 dollars)		
Direct Costs	6,747	
Indirect Costs	<u>5,221</u>	
Total	11,968	
2. Contingency	2,992	
3. Owner's Cost	1,197	
4. Replacement Power Costs	<u> -</u>	
Total	16,157	
5. Annual Capital Charges		2,747
6. O&M Costs (1/85 dollars)	114	
7. Levelized Annual O&M Costs		<u>168</u>
8. Total Annual Costs		2,915

TABLE 4.1-2
CAPITAL COSTS FOR
ALTERNATIVE 1 - GENERIC BASIS
(\$ in Thousands, 1/85 Cost Base)

<u>Cost Component</u>	<u>Labor</u>		<u>Materials</u>	<u>Total</u>
	<u>Manhour</u>	<u>Amount</u>		
MOD 106	1,400	40	2	42
MOD 107	1,170	33	1	34
MOD 112	460	13	1	14
MOD 303	38,220	1,080	775	1,855
MOD 801	14,150	413	60	473
MOD 804	72,420	2,134	531	2,665
MOD 805	42,170	1,247	98	1,345
MOD 811	<u>10,370</u>	<u>303</u>	<u>16</u>	<u>319</u>
Total Direct	180,360	5,263	1,484	6,747
Temporary Constr.			3,872	3,872
Engrg. & Constr. Mgt.			<u>1,349</u>	<u>1,349</u>
Total Indirect			5,221	5,221
TOTAL BASE CAPITAL	180,360	5,263	6,705	11,968

TABLE 4.1-3
TOTAL EVALUATED COSTS FOR
ALTERNATIVE 1 - LOCAL BASIS
(Dollars in Thousands)

1. Base Capital Costs (1/85 dollars)		
Direct Costs	5,973	
Indirect Costs	<u>4,618</u>	
Total	10,591	
2. Contingency	2,648	
3. Owner's Cost	1,059	
4. Replacement Power Costs	<u>-</u>	
Total	14,298	
5. Annual Capital Charges		2,288
6. O&M Costs (1/85 dollars)	80	
7. Levelized Annual O&M Costs		<u>135</u>
8. Total Annual Costs		2,423

TABLE 4.1-4
CAPITAL COSTS FOR
ALTERNATIVE 1 - LOCAL BASIS
(\$ in Thousands, 1/85 Cost Base)

<u>Cost Component</u>	<u>Labor</u>		<u>Materials</u>	<u>Total</u>
	<u>Manhour</u>	<u>Amount</u>		
MOD 106	1,400	39	2	41
MOD 107	1,170	32	1	33
MOD 112	460	12	1	13
MOD 303	38,220	977	775	1,752
MOD 801	14,150	351	60	411
MOD 804	72,420	1,781	531	2,312
MOD 805	42,170	1,040	98	1,138
MOD 811	<u>10,370</u>	<u>257</u>	<u>16</u>	<u>273</u>
Total Direct	180,360	4,489	1,484	5,973
Temporary Constr.			3,423	3,423
Engrg. & Constr. Mgt.			<u>1,195</u>	<u>1,195</u>
Total Indirect			4,618	4,618
TOTAL BASE CAPITAL	180,360	4,489	6,102	10,591

4.2 Alternative 2

4.2.1 Operations and Maintenance

Alternative 2 consists entirely of Modification 812 which is the Add-on Decay Heat Removal System. Being a major system housed in its own structures, it is expected to have considerably more impact on operations and maintenance than will any of the other alternatives.

Since the add-on structures will not normally need to be continuously manned by operations personnel, no additional operators are expected to be needed. However, this system will require additional operator training, and additional operator actions during plant operations. The operator will need a good understanding of the interactions of this system with existing decay heat removal trains. In particular, he should know the priority of initiation of the existing and add-on decay heat removal trains and be aware of and be able to interpret indications of each train's performance.

Particular attention will need to be given to this in the design process to ensure that the design of the instrumentation and controls makes the transition to the decay heat removal mode as simple and unambiguous as practical. During normal operations, the operator will need to do additional line-up checks to ensure that the system is ready to work in the intended manner when it is called upon to do so.

Considering that the reason for adding this system is to bring the plant to a minimum desired level of safety, it is expected that any failures within the add-on system will constitute a limiting condition for operation. Thus, existence of this system will result in some reduction in overall plant availability.

In-service inspection of all ASME Code, Section III, Class 1, 2 and 3 components in accordance with ASME Code, Section XI will be required. In addition, periodic functional testing of pumps and valves will be required to ensure operability. Also the diesel generator must be tested in accordance with the requirements of Regulatory Guide 1.108, "Periodic Testing of Diesel Generator Units Used as On-site Electrical Power Systems at Nuclear Power Plants". This will require weekly operation at operating temperature as well as monthly operation at full-load conditions.

In addition, the substantial amount of added equipment and structures will require considerable preventive and corrective maintenance as well as general preservation and cleanup activities.

Operations and maintenance hours were not estimated in detail due to the scope of activities required. Rather O&M costs were estimated as 1% of the direct capital costs

which is at the low end of the range of what might be predicted using the methods of References 3 and 6. The low end of the range was selected because some components of cost will not increase (e.g., number of security guards and operators) and the unit is seeing limited duty since it is normally in standby with required operation expected to be infrequent.

4.2.2 Total Evaluated Costs

Total evaluated costs estimated to result from this alternative are as shown in Tables 4.2-1 and 4.2-3 for the generic and local cases, respectively. Associated capital costs showing a breakdown of direct costs by modification and total indirect costs for the entire alternative are shown in Tables 4.2-2 and 4.2-4. Owner's costs, contingency, replacement power, annual capital charges, and O&M costs were calculated as described in Sections 3.3.2 and 4.1.1.

TABLE 4.2-1
TOTAL EVALUATED COSTS FOR
ALTERNATIVE 2 - GENERIC BASIS
(Dollars in Thousands)

1. Base Capital Costs (1/85 dollars)		
Direct Costs	20,609	
Indirect Costs	<u>17,671</u>	
Total	38,280	
2. Contingency	9,570	
3. Owner's Cost	3,828	
4. Escalation and AFUDC	7,607	
5. Replacement Power Costs	<u>-</u>	
Total	59,285	
6. Annual Capital Charges		10,078
7. O&M Costs (1/85 dollars)	383	
8. Levelized Annual O&M Costs		<u>563</u>
9. Total Annual Costs		10,641

TABLE 4.2-2
CAPITAL COSTS FOR
ALTERNATIVE 2 - GENERIC BASIS
(\$ in Thousands, 1/85 Cost Base)

<u>Cost Component</u>	<u>Labor</u>		<u>Materials</u>	<u>Subcontracts</u>	<u>Total</u>
	<u>Manhour</u>	<u>Amount</u>			
MOD 812	<u>418,850</u>	<u>11,501</u>	<u>8,792</u>	<u>316</u>	<u>20,609</u>
Total Direct	418,850	11,501	8,792	316	20,609
Temporary Constr.			11,694		11,694
Engrg. & Constr. Mgt.			<u>5,977</u>		<u>5,977</u>
Total Indirect			17,671		17,671
TOTAL BASE CAPITAL	418,850	11,501	37,964	316	38,280

TABLE 4.2-3
TOTAL EVALUATED COSTS FOR
ALTERNATIVE 2 - LOCAL BASIS
(Dollars in Thousands)

1. Base Capital Costs (1/85 dollars)		
Direct Costs	19,025	
Indirect Costs	<u>16,293</u>	
Total	35,318	
2. Contingency	8,829	
3. Owner's Cost	3,532	
4. Escalation and AFUDC	7,018	
5. Replacement Power Costs	<u>-</u>	
Total	54,697	
6. Annual Capital Charges		8,752
7. O&M Costs (1/85 dollars)	353	
8. Levelized Annual O&M Costs		<u>597</u>
9. Total Annual Costs		9,349

TABLE 4.2-4
CAPITAL COSTS FOR
ALTERNATIVE 2 - LOCAL BASIS
(\$ in Thousands, 1/85 Cost Base)

<u>Cost Component</u>	<u>Labor</u>		<u>Materials</u>	<u>Subcontracts</u>	<u>Total</u>
	<u>Manhour</u>	<u>Amount</u>			
MOD 812	<u>418,850</u>	<u>9,917</u>	<u>8,792</u>	<u>316</u>	<u>19,025</u>
Total Direct	418,850	9,917	8,792	316	19,025
Temporary Constr.			10,776		10,776
Engrg. & Constr. Mgt.			<u>5,517</u>		<u>5,517</u>
Total Indirect			16,293		16,293
TOTAL BASE CAPITAL	418,850	9,917	35,002	316	35,315

The construction schedule for the add-on system is estimated to be roughly 18 months to 2 years from ground breaking to the the completion of operational testing. It is assumed that at least six months of engineering must be completed before construction could begin. Much of the construction can proceed independent of plant operations; however, tie-ins to the feedwater and Reactor Coolant Systems can only be done with the plant shut down. The tie-ins can be completed in a single 60-day refueling shutdown. Thus, shift differential costs have not been included in the capital cost estimate and there are no replacement power costs.

Escalation and AFUDC costs were computed assuming the bulk of the expenditures would occur during the period January 1986 to June 1987 with a linear cumulative cash flow.

4.2.3 Radiation Exposure

This alternative includes only Modification 812. This modification requires some work in radioactive areas to route piping through the auxiliary building into the pipe and valve room to make the tie-in to the Chemical Volume and Control system piping. The total estimated exposure for this work is 20 man-rem.

5.0 CONCLUSIONS

Table 5.0-1 provides a summary and comparison of the impact assessment for each of the alternatives. This table shows the results for the generic cases only. There is no difference in ranking among alternatives for the generic and local cases, only a reduction in the costs since the local labor rates are lower.

It is evident from the table that Alternative 1 which includes only changes to existing systems or structures, can be achieved at a more modest cost and have less impact on plant operations and maintenance. This alternative can be also achieved with a negligible radiological exposure. The remaining alternative, by adding new structures and a considerable amount of additional equipment, has a major capital cost impact in the range of four times that of Alternative 1. The operational and maintenance impact is also much more significant and the radiological exposure is greater.

Since both of the alternatives involve addition of some active equipment which is required to bring the plant to some prescribed level of safety, it is presumed that the unavailability of this equipment, once installed, will be a limiting condition for operations. Thus, it is expected that all alternatives will serve to reduce plant availability somewhat. Although that reduction was not quantified, it is not expected to be large.

It should be recognized that there is a non-negligible degree of uncertainty in this work. Typically, in modification work, detailed design and a careful system walkdown will identify design features which were not

anticipated. Similarly, such an investigation will permit a better feel for application of the "complexity factors" used in cost estimating for the specific modification. Limited budget and time constraints did not permit each modification to be investigated in such detail. The gap between Alternatives 1 and 2 is so large that these uncertainties would not affect their relative ranking.

TABLE 5.0-1
ALTERNATIVES IMPACT SUMMARY
ARKANSAS NUCLEAR ONE DECAY HEAT ALTERNATIVES

<u>Alternative</u>	<u>Annualized Capital Costs</u> (Inflated \$ x 10 ³)	<u>Operations</u>	<u>Maintenance</u>	<u>Radiation Exposure</u> (Man-Rem)
1	2,915	Minimal-some operator training required but day-to day operations will not be significantly affected.	Minimal-requires less than 225 mh/yr primarily for battery testing and maintenance.	0
2	10,641	Moderate - requires additional line-up checks and significant operator training	Significant - will cost on the order of half a million dollars a year to maintain the added equipment and structures.	20

6.0 REFERENCES

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5. Replacement Energy Costs for Nuclear Electricity-Generating Units in the United States, Argonne National Laboratories, J. C. VanKuiken, et.al. (NUREG/CR 4012, October 1984).
6. Non-Fuel Operations and Maintenance Costs for Large Steam-Electric Power Plants, Oak Ridge National Laboratories, M. L. Myers, et.al. (1982)
7. Estimating Ground Rules for the HTGR Program, Gas-Cooled Reactor Associates (GCRA 83-008, February, 1984).
8. Energy Economic Data Base (EEDB) Program, Phase VII Update (1984) by United Engineers & Constructors Inc. for Oak Ridge National Laboratories (to be issued - DOE release pending).

ATTACHMENT A

MODIFICATION DESIGN REPORT

for

MODIFICATIONS TO THE AUXILIARY FEEDWATER PUMP

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DESIGN REPORT

MODIFICATIONS TO THE AUXILIARY FEEDWATER PUMP

1.0 FUNCTION AND DESIGN REQUIREMENTS

1.1 Function and Purpose of the Modification

Prevent complete loss of auxiliary feedwater due to a local fault of the turbine driven auxiliary feedwater pumps, local fault of the batteries or a combination of the two.

1.2 Design Requirements and Criteria

1.2.1 Performance

The existing motor driven auxiliary feedwater pump will be piped to the existing Unit 1 Condensate Storage Tank and provided with a Class 1E power source.

1.2.2 Safety/Regulatory/Availability

Safety Class 3 (Mechanical)
Safety Class 1E (Electrical)
Availability shall be continuous.

1.2.3 Environmental

Location - Indoors
Temperature - 55-120°F
Seismic - Equipment and piping shall be seismic Category I.

1.2.4 Installation

Pumps, motors and valves shall be accessible for inspection and maintenance.

1.2.5 Controls

Operating status of pump and position of the Mov.'VI' shall be indicated locally and in the Main Control Room. Operation of the pump shall be manually controlled from the Main Control Room and from the local control panel. Operation of the Mov.'VI' shall be from the local panel.

The pump shall be tied as it is currently; to the plant A2 bus. Transfer switches shall be installed to cross tie all the batteries to the existing Auxiliary Feedwater control valves.

1.2.6 Testing

Piping and valves shall be hydrostatically tested at installation. Electrical system shall be tested after installation in accordance with 10CFRS Appendix A.

1.3 Design Codes

ASME B&PV Section III, Subsection ND
Class 3 Components
IEEE Standard 308

1.4 References

- Bechtel Dwg. 6600-M-5 - Equipment Location - Ground Floor Plan
Rev.17, 8/2/85
- Bechtel Dwg. 6600-M-6 - Equipment Location - Plan Below Grade
Rev.13, 11/7/84
- AP&L Dwg. M-204 - Piping & Instrument Diagram Emergency
Feedwater - Rev. N, 7/19/84

2.0 DESIGN DESCRIPTION

2.1 Modification Configuration

The existing electric motor driven auxiliary feedwater pump will be connected to the Unit 1 Condensate Storage Tank.

The pump will remain piped as it is now and will also be tied to the condensate storage tank (suction) and the auxiliary feedwater supply header. The arrangement of the piping system is shown in the schematic diagram of Figure 1.

The pump is powered from the plant A2 bus and will be able to be used either as a start-up feedwater pump or an auxiliary feedwater pump.

New transfer switches will provide the capability of powering the existing Main Feedwater startup control valves (CV 2623 and 2673) and the feedwater header cross tie valve, CV 2827, from a Class 1E power source.

2.2 Operation/Performance

The auxiliary feedwater pump is normally aligned to the Main Feedwater pump header and discharge to the normal feedwater supply header. This alignment allows use of the pump as a start-up feedwater pump during normal plant operation.

When required, the pump can be used as an additional emergency feedwater pump by aligning the pump suction to the condensate storage tank. This realignment is accomplished by repositioning a motor-operated valve (V-1) from a local control panel. The existing main feedwater control valves are used to control flow to the steam generators.

Power to the pump is normally provided by the station A2 bus (Figure 2). If off-site power is not available, the pump can be powered by cross tying the A2 bus to the diesel generators. This will require stripping the other loads on the A2 bus.

2.3 Controls

The position of valve V-1 will be indicated by open/closed lights on the local panel and in the main control room. Indicating lights on a local panel and in the main control room will indicate when the pump is running. The pump is controlled either from the local panel or the main control room (see Figure 2).

Discharge valves CV-2623, 2673 and 2827 are provided with two sources power through automatic transfer switches. Operation of the automatic transfer switch is annunciated in the control room (see Figure 3).

3.0 COMPONENT DESIGN BASIS

3.1 Pump and Motor

The existing pumps are horizontal centrifugal, multi-stage designed for demineralized, treated water. The nameplate motor rating is 400 HP.

3.2 Piping

Design, fabrication and erection of all piping in this modification will be in accordance with ASME-B&PV Section III, Subsection ND. Piping and valves will be designed and supported for local seismic loads.

Material	- Carbon steel
Pressure	- 1200 psi
Temperature	- 100°F
Fluid	- Demineralized, treated water

3.3 Valves

V-1,	- Motor operated gate
Size	- 8"
Material	- Cast carbon steel
Pressure	- 150 psi
Temperature	- 100°F
Type	- Double disc gate, butt welded ends
Accessories	- Open and closed limit switches.

3.4 Electrical Components

- One Local Control Panel (18' x 12' x 8") (stanchion mounted) with 2-light/switch assemblies
- One Control Room Status Light
- 1 - Size 1 Reversible Starters with breaker and overload relays for mounting in existing Non-vital MCC.
- 2 - Automatic Transfer Switches, w/contacts for control room annun.
- Conduit - 1400'
- Power Cable - 1000'
- Control Cable - 1800'

4.0 INTERFACE REQUIREMENTS

- Utilize existing auxiliary feedwater pump supply and discharge headers.
- Utilize existing auxiliary feedwater pump and discharge valves.
- Provide power supply to the motor drive from the A2 bus.
- Provide operational control of the pump from a local panel and from the main control room.
- Provide valve position indication and control for valve V1 locally and in the main control room.
- Provide pump motor operating status locally and in the main control room.

5.0 SCOPE OF WORK

5.1 Installation

- Erect suction piping together with isolation valve and seismic supports.
- Install conduit with seismic supports at 10' spacing.

- Install cables for pump motor controls.
- Install local control panel with seismic supports.
- Install control switches and lights in control room.
- Install automatic transfer switches in control room.
- Install starter in 480V MCC.
- Test complete electrical system.
- Perform hydrostatic test of piping and operational tests of pumps, valves and controls.

6.0 SEQUENCE OF ACTIVITIES AND SCHEDULE

6.1 Activities Before Plant Shutdown

- 6.1.1 Prefabricate longest practical sections of new piping.
- 6.1.2 Erect new piping and install new valve.
- 6.1.3 Install local control panel.
- 6.1.4 Install conduits and wiring for power supply to motor operated valve.
- 6.1.5 Erect conduits and wiring for control cables between local control panel, MCC/Switchgear and main control room except control room terminations.

	<u>Start</u>	<u>Shutdown</u>
6.1.1	XXXX	
6.1.2	XXX	
6.1.3	XX	
6.1.4	XX	
6.1.5	XX	

6.2 Activities After Plant Shutdown

- 6.2.1 Isolate and drain sections of existing headers containing the points of interface with this modification.
- 6.2.2 Make required cuts into existing headers and install saddles, weldolets or tees, as required.
- 6.2.3 Complete connections between suction and discharge piping of the existing headers.
- 6.2.4 Perform required nondestructive examination of all field welds.
- 6.2.5 Perform hydrostatic test of piping.
- 6.2.6 Install control switches and lights in main control board. Install automatic transfer switches. Install internal panel wiring. Install MCC compartments.

Switchgear and MCC compartments.
- 6.2.7 Complete valve motor power supply wiring.
- 6.2.8 Complete control wiring.
- 6.2.9 Perform pump operational test.
- 6.2.10 Test controls system.

Shutdown	
6.2.1	XXX
6.2.2	XXX
6.2.3	XXX
6.2.4	XX
6.2.5	XXX
6.2.6	XXX
6.2.7	XX
6.2.8	XX
6.2.9	XX
6.2.10	XXXX

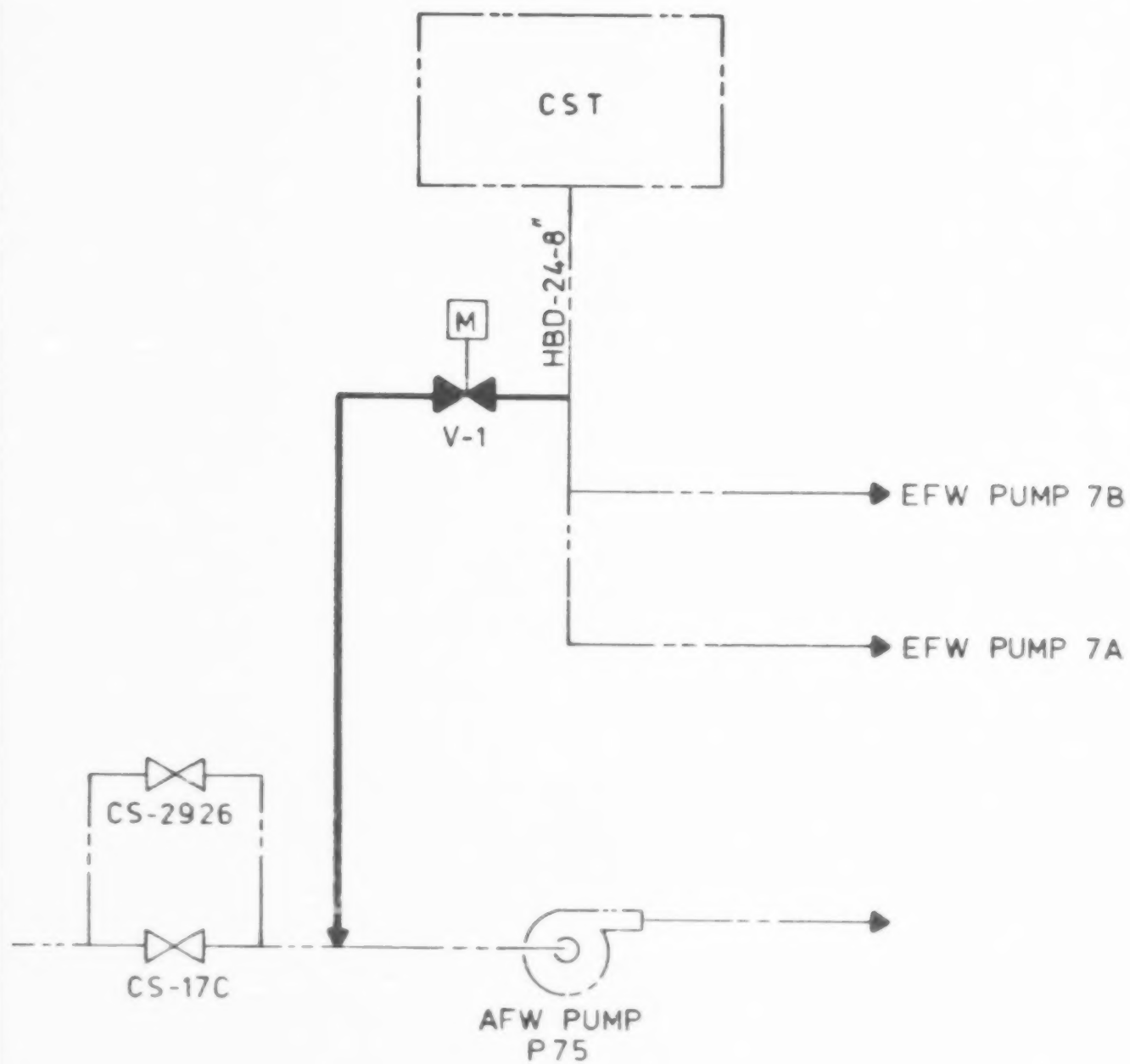
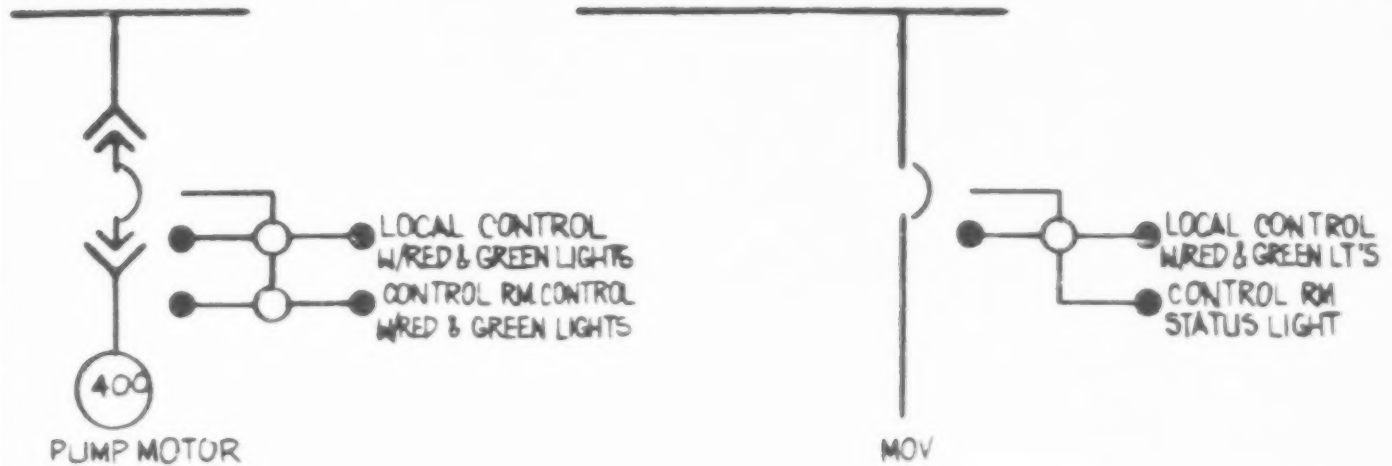


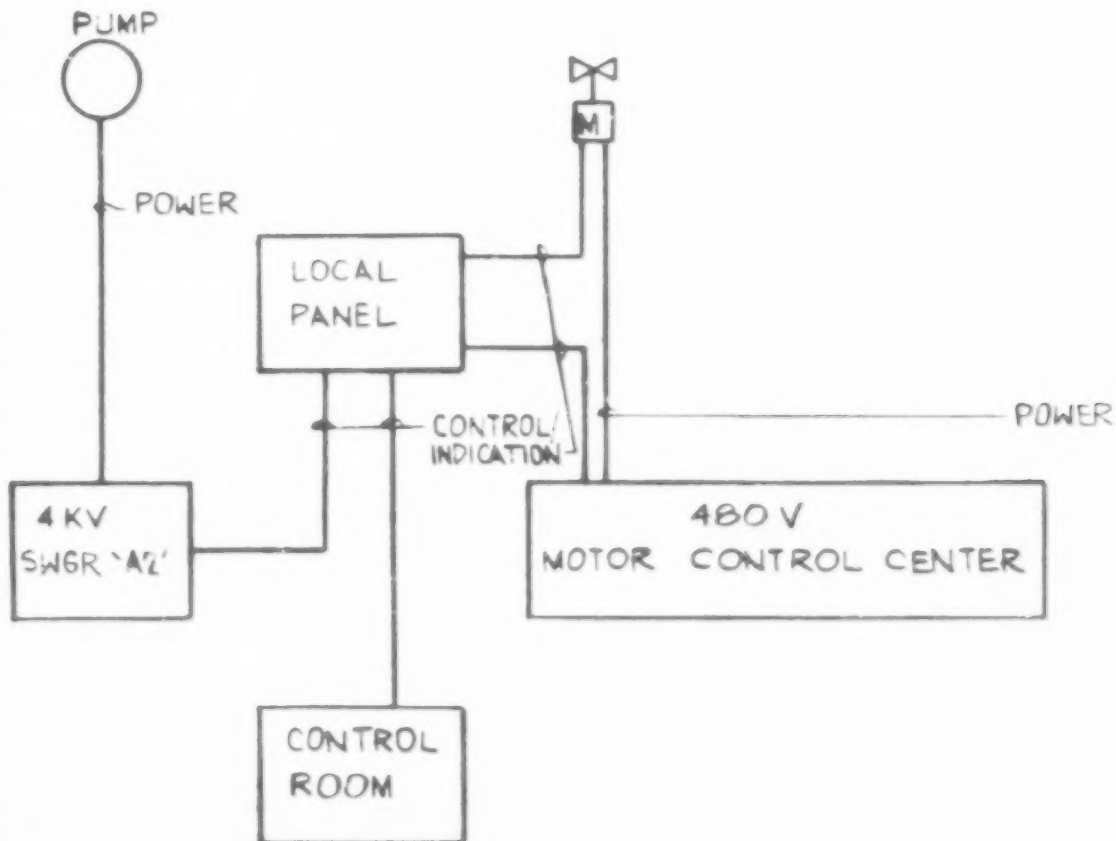
FIGURE 1

4KV SWGR 'A2'

480V MCC 'NON VITAL'

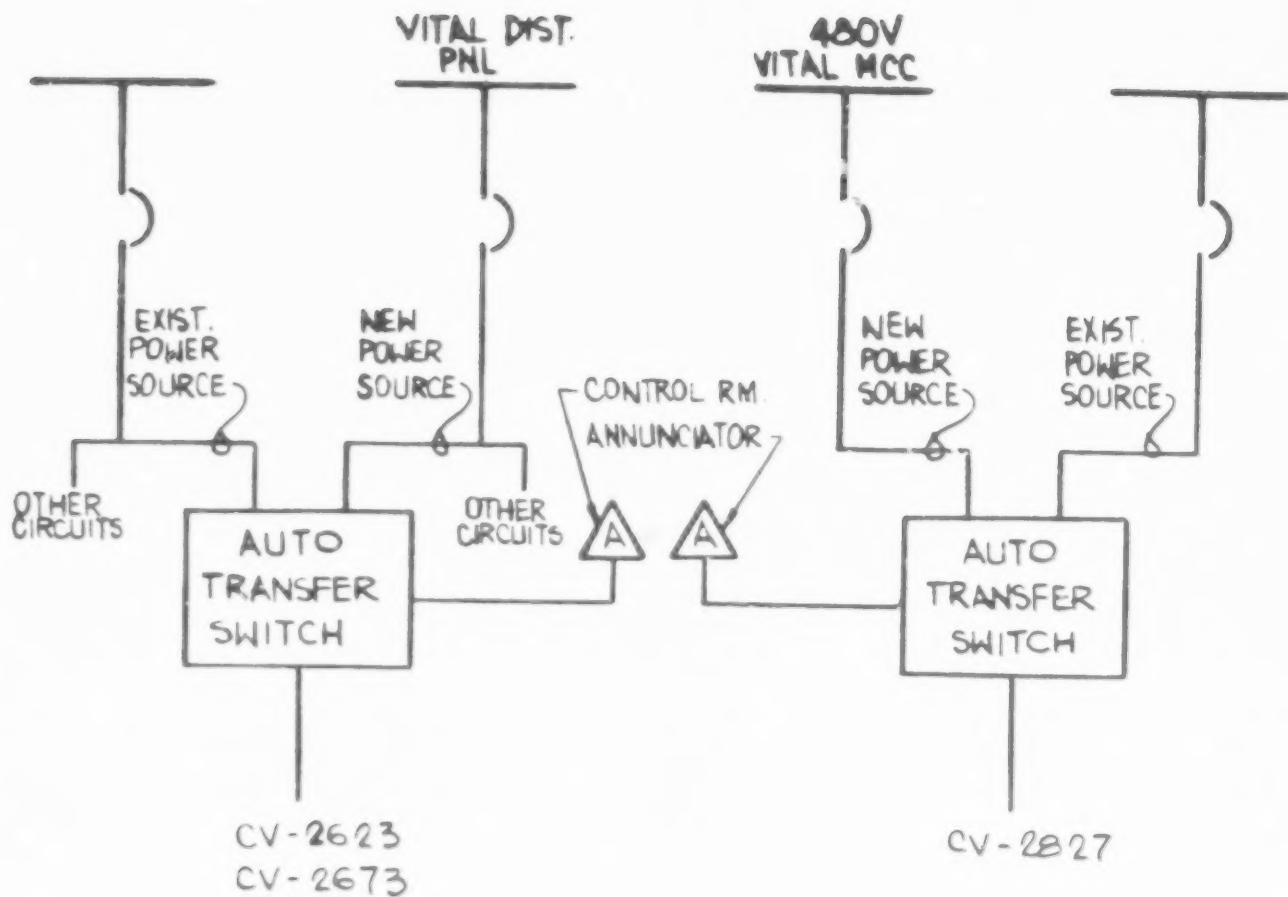


SINGLE LINE DIAGRAM



BLOCK DIAGRAM

FIGURE 2



SINGLE LINE DIAGRAM

FIGURE 3

ATTACHMENT B

MODIFICATION DESIGN REPORT

for

ADDITION OF A

TURBINE DRIVEN GENERATOR

FOR EMERGENCY LOADS

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DESIGN REPORT
for
TURBINE DRIVEN GENERATOR

1.0 FUNCTION AND DESIGN REQUIREMENTS

1.1 Function and Purpose of Modification

The purpose of this modification is to reduce the plant vulnerability to a long term station blackout by providing a back-up power supply to the station battery chargers.

1.2 Design Requirements and Criteria

1.2.1 Performance/Criteria

The installed system shall be capable of supplying 500 kW to provide an independent power source for the battery chargers, 1 400 HP auxiliary feed water pump, control power and vital instrumentation.

1.2.2 Safety/Regulatory/Availability

Safety Class	- 3 (Mechanical)
Class	- 1E (Electrical)
Regulatory	- ASME/ANSI
Availability	- Continuous operation above cold shutdown

1.2.3 Environmental

Location	- Indoors
Temperature	- 20°F - 110°F
Seismic	- Category I
Radiation	- Low

1.2.4 Installation

Equipment for this modification shall be located outside the existing AFW pump room, on the turbine operating floor and in the control room and shall be accessible for inspection and maintenance.

1.2.5 Controls

All controls and switchgear for this modification shall be local/ manually operated. Remote indication of transfer switch positions shall be provided in the central control room.

1.2.6 Testing

Hydrostatic testing of piping and valves is required after installation. The electrical system is to be tested in accordance with 10CFR50, Appendix A.

1.3 Design Codes

ASME III, Subsection ND, Class 3 Component
10CFR50, Appendix A
IEEE-323-1974
IEEE-344-1975

1.4 References

* Bechtel Dwg. No. 6600-M-3, Plan Operating Floor Rev.15 - 9/7/83,
6600-M-4, Plan Intermediate Floor Rev.13 10/3/84, 6600-M-6, Plan
Below Grade Rev.11 - 3/6/82.

2.0 DESIGN DESCRIPTION

2.1 Modification Configuration

The turbine generator assembly for the proposed modification will be located outside of the auxiliary feedwater pump room north of existing pump P7B. Steam to drive the turbine will be supplied via a 4" pipeline connected to the existing main steam line that feeds pump P7A. A 4" globe valve in the new line provides manual isolation of the turbine. Turbine exhaust will be through a 12" pipe which discharges to the atmosphere. A conceptual diagram of the piping connections and valves is provided in Figure 1.

The 500 kW generator will power a 480 volt substation (via the existing 4160 Vac Bus A2) located near the turbine generator assembly. This substation then feeds the 480 volt MCC and the power and control transfer switches which are located in the control room. The battery chargers are provided with a separate manually operated transfer switch. Local and remote switch position indicators are provided for each transfer switch. Conceptual single line diagrams for this modification are shown in Figures 2.

2.2 Operation/Performance

During normal plant operation, the turbine driven generator will be isolated from the main steam supply by means of a globe valve, and all equipment and circuits will be fed from the existing 4160 volt buses. If a station blackout occurs (loss of off-site power and D.G. failure), an operator will open the new manual globe valve allowing main steam to enter the turbine steam chest which contains

the turbine governor and trip valves. After the turbine generator has achieved operating speed, the operator will transfer the power feed for the battery chargers from the normal 480 volt bus to the new source to allow continued operation of the vital instrumentation.

2.3 Controls

The turbine generator is a constant speed device and is controlled by a skid-mounted governor and control valve. Operation of this unit simply requires opening a manual valve located near the turbine skid. Power transfer switches are also controlled locally with position indication (normal feed/emergency feed) provided locally and in the control room.

3.0 COMPONENT DESIGN BASIS

3.1 Turbine

Type	- Horizontal axial flow, noncondensing unit
Horsepower	- 750
Steam Flow	- 30,000 lb/hr
Seismic Category	- I

3.2 Generator

Rotating (kW)	- 500
Type	- Brushless
Seismic Category	- I

3.3 Piping

Piping shall be Seismic Category I, designed and built in accordance with ANSI/ASME BPV-III-1-ND.

Material	- Carbon steel
Design Pressure	- 1085 psig (inlet)/ 50 psig (exhaust)
Design Temperature	- 600°F (inlet/ 400°F (exhaust)

3.4 Valves (1 each)

A) 4" Globe

Type	- Manual
Material	- A216-WCB
Pressure Class	- 600 lb
Max. Press. Diff.	- 1085 psig

B) 1" Globe

Type	- Manual
Material	- A216-WCB
Pressure Class	- 600 lb
Max. Press. Diff.	- 1085 psig

C) 12" Check

Type	- Swing Check/Butt weld ends
Material	- A216-WCB (Carbon steel)
Pressure Class	- 150 lb

3.5 ELECTRICAL COMPONENTS

Qty.

- 1 Power Transfer Switch (4160V, 100 Amps., 3 Ø) with an auxiliary switch for position indication.
- 2 Power Transfer Switch (480V, 200 Amps., 3 Ø) with an auxiliary switch for position indication.
- 1 Control Transfer Switch (125 VDC, 20 Amps., 4 gang - 2 pole).
- 1 MCC Stack with three (3) fused disconnect switches rated (1) 100A, (2) 225A Amp. (3 Ø) and control power transformer for indicating lights.
- 1 30 Amp. Battery Charger 480 V AC Input 125 V DC output.
- 1 Battery 125 V DC 59 Cell 150 A.H.
- 1 125 V DC power panel w/2 pole 100 Amp. Main and Sub-feed lugs for battery connection.
 - 5 Branch Circuit Capacity with 1 50 Amp. &
 - 4 20 Amp. 2 pole breakers
- 1500' Power Cable
- 1700' Control Cable
- 900' Conduit with seismic supports at 10' spacings

4.0 INTERFACE REQUIREMENTS

Normal 480 volt power supply to the battery chargers will be fed through the new transfer switches and will require cable routing.

The main steam piping tie-in is classified as Seismic Category I, and will require reanalysis of existing piping and support design.

- 5.0 SCOPE OF WORK
- 5.1 Installation
 - 5.1.1 Erect a Seismic Category I concrete pad for the turbine generator assembly.
 - 5.1.2 Erect a 4" Schedule 80 pipe from the existing main steam to AFW header to the new turbine generator.
 - 5.1.3 Erect a 12" Schedule 40 pipe from the new turbine exhaust to the AFW pump turbine exhaust.
 - 5.1.4 Install Seismic Category I supports for all newly erected piping.
 - 5.1.5 Install equipment power and control cabling and conduit from the generator to the transfer switch and from the transfer switch to the 4160 VAC switchgear, 480 VAC MCC to transfer switches and new battery charger from existing battery chargers to transfer switches.
 - 5.1.6 Install control cables from the transfer switches to the control room for remote indication.
- 5.2 Rip-out and Relocation
 - 5.2.1 Remove existing power cables from the 480 volt bus to the battery chargers.

6.0 SEQUENCE OF ACTIVITIES AND SCHEDULE

6.1 Activities Before Plant Shutdown

- 6.1.1 Core bore through walls and install 4" and 12" sleeves as required for turbine inlet and exhaust piping.
- 6.1.2 Pour concrete pad for turbine generator.
- 6.1.3 Install and align turbine generator assembly.
- 6.1.4 Erect all 12" turbine exhaust piping.
- 6.1.5 Erect 4" steam supply piping up to the main steam connection point.
- 6.1.6 Install manual transfer switches, new MCC section, battery charger, battery and distribution panel.
- 6.1.7 Install in-connecting power feed (conduit and cable) from the generator to transfer switch, new MCC stack to transfer switches and new battery charger, from new battery charger to distribution panel and to battery.
- 6.1.8 Install power feed (conduit and cable) from the transfer switches to the battery chargers (D02, D04).
- 6.1.9 Install remote indicators, control wiring (conduit and cable & DC panel circuits).
- 6.1.10 Terminate cables to non-energised equipment.
- 6.1.11 Install power feed (conduit and cable) from the existing 4160 volt bus to the transfer switch except for the final tie-in at the bus.
- 6.1.12 Seal spaces between 4" and 12" pipes and wall sleeves.

	<u>Start</u>	<u>Shutdown</u>
6.1.1	X	
6.1.2	XX	
6.1.3	XXX	
6.1.4	XXX	
6.1.5	XX	
6.1.6	XXX	
6.1.7	XX	
6.1.8	XX	
6.1.9	XX	
6.1.10	XX	
6.1.11	XX	
6.1.12	XX	

6.2 Activities After Plant Shutdown

- 6.2.1 Complete erection of 4" piping and connect to the existing main steam line.
- 6.2.2 Perform hydrostatic test of newly installed piping.
- 6.2.3 Connect newly installed power cables to the battery chargers, 4160V bus and control cables. Rip out the existing cables to this equipment.
- 6.2.4 Perform function check of system.

Shutdown

- 6.2.1 XX
- 6.2.2 X
- 6.2.3 XXX
- 6.2.4 XX

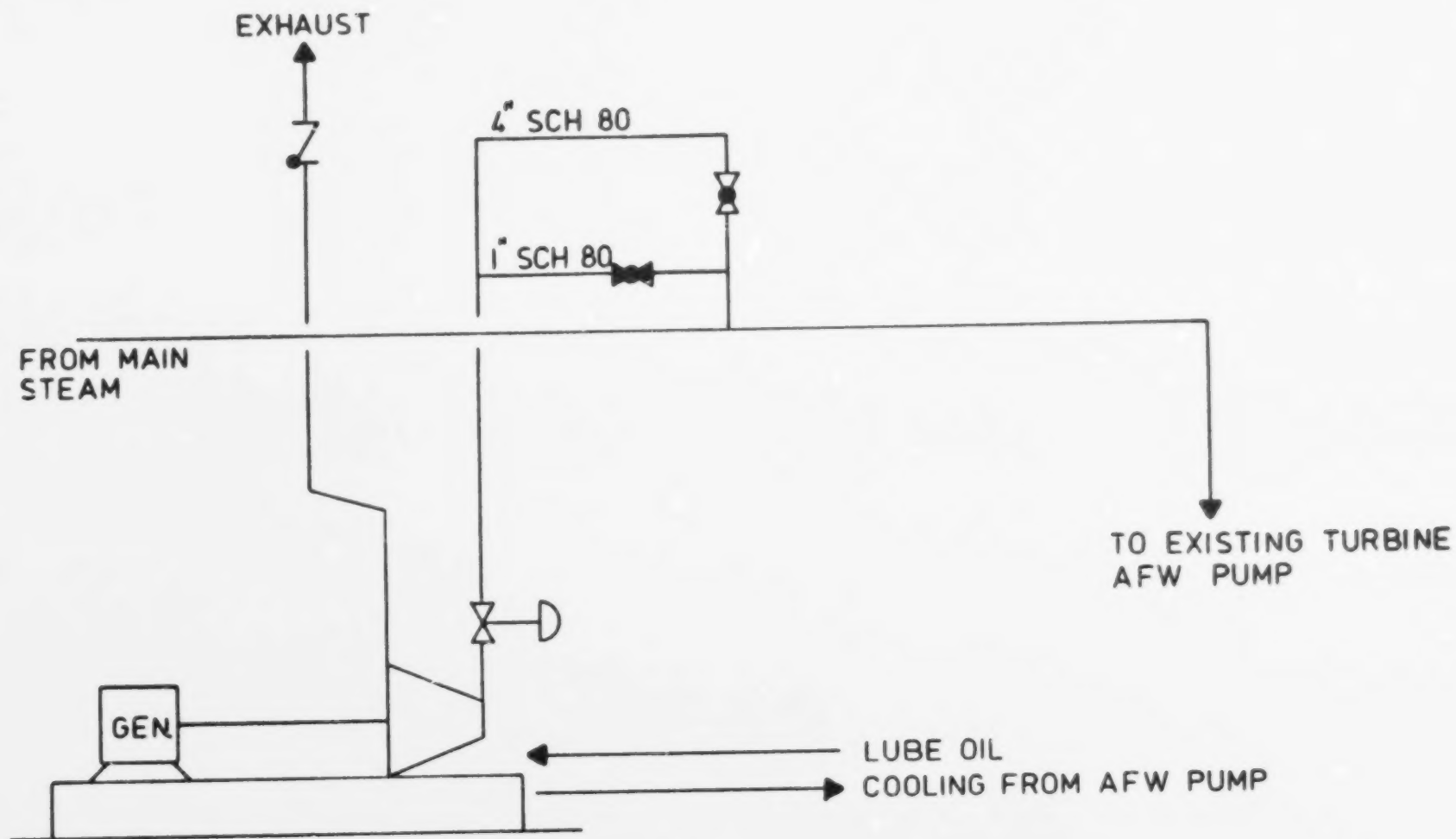


FIGURE 1
TURBINE DRIVEN GENERATOR
FLOW DIAGRAM

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J-67

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ATTACHMENT C

MODIFICATION DESIGN REPORT

for

ADDITION OF A THIRD

DECAY HEAT REMOVAL PUMP

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MODIFICATION DESIGN REPORT

for

ADDITION OF A THIRD

DECAY HEAT REMOVAL PUMP

1.0 FUNCTIONS AND DESIGN REQUIREMENTS

1.1 Function and Purpose of the Modification

The function of this modification is to reduce the plant vulnerability to a loss of the decay heat removal (DHR) system due to pump failure.

1.2 Requirements

1.2.1 Performance

The new pump, piping and valves shall be capable of providing full decay heat removal flow and head requirements.

The pump shall be capable of supplying flow to either DHR train. (Figure 1).

1.2.2 Safety/Regulatory/Availability

Class - 1E (Electrical)
Safety Class - 2 (Mechanical)
Regulatory - ASME/ANSI
Availability - Continuous

1.2.3 Environmental

Location - Indoors
Temperature - 20°F to 110°F
Seismic - Category I
Radiation - High

1.2.4 Installation

Equipment shall be located outside the existing ESF B Train pump room and shall be accessible for maintenance and inspection. (Figure 2)

1.2.5 Control

Alignment of the new pump is a manual operation. Local pump controls are provided with status lights in the control room.

The pump shall be capable of being powered from either Class 1E distribution system.

1.2.6 Testing

All piping shall be hydrostatically tested after installation. The new pump shall be periodically tested in accordance with the requirements of the existing DHR pumps.

The valve encapsulation structures shall be subject to periodic leakage testing in accordance with the existing commitments for the containment.

Electrical equipment shall be tested after installation in accordance with 10CFR50, Appendix A.

1.3 Design Codes

ASME III Subsection NC Class 2

1.4 Reference

- * Bechtel Dwg. No. 6600-M-11 Equipment Location - Misc. Plans & Sections EL. 317'-0"; Rev. 10, P2, 4/2/82.

2.0 DESIGN DESCRIPTION

2.1 Modification Configuration

This modification consists of adding a third 100% DHR pump in parallel with the existing pumps (see Figure 1). The pump will be located outside the existing ESF B Train pump room on elevation 317'-0" and will be capable of replacing either existing DHR pump.

2.2 Operation/Performance

During normal plant operation, the new pump will be isolated by locked closed valves on the suction and discharge headers to maintain separation of the existing DHR trains.

The new pump will be utilized only in the event of a loss of one of the existing pumps.

2.3 Controls

The new DHR pump is capable of being powered by either Class 1E distribution system via a new power transfer switch (see Figure 3).

The pump is manually aligned to either DHR train and operated locally with status indication in the main control room.

3.0 COMPONENT DESIGN BASIS

3.1 Mechanical Equipment

- ° RHR Pump
 - 2500 GPM @ 400 Ft. TDH
 - 350 HP
 - ASME Seciton III Subsection NC
- ° Piping
 - Material: Stainless Steel
 - Pressure: 600 psig
 - Temperature: 400 °F
 - Fluid: Water
- ° Valves
 - Manual OS&Y Gate 2-10", 2-8"
 - 600 Lb. Rating
 - Stainless Steel
- ° Valve Enclosure and Guard Pipe
 - Material: Stainless Steel
 - Pressure: 100 psig
 - Temperature: 300°F

3.2 Electrical Equipment

- ° Power Transfer Switch (4160V, 100 Amps, 3 phase) with an auxiliary switch for position indication
- ° 4160V Switchgear Breakers - 100 Amp with control switch and indicating lights
- ° Annunciator - 6 window (2 x 3) 125VDC
- ° Power Cable - #6 AWG 5KV
- ° Conduit with seismic supports @ 10' spacing.

4.0 INTERFACE REQUIREMENTS

4.1 Supporting Systems

- Power Supply from each 480V 1E distribution system
- Room is required for three status lights on the DHR panel in the main control room

5.0 SCOPE OF WORK

5.1 Installation

- Construct new pump pedestal and block wall
- Install new pump
- Install new pipe, valves and supports (except tie-ins to existing system)
- Pull cable
- Install local control panel
- Install status lights on the main panel
- Make tie-ins to existing piping and electrical systems
- Perform hydrostatic test
- Perform functional test

6.0 SEQUENCE OF ACTIVITIES AND SCHEDULE

6.1 Activities Before Plant Shutdown

	<u>Start</u>	<u>Shutdown</u>
• Construct pump pedestal and block wall	XXX	
• Prefab piping and valves	XXX	
• Install conduit & supports	XX	
• Pull cable	XX	
• Install new pump and motor		XX
• Install local panel		XX
• Install piping, valves and supports	XXX	

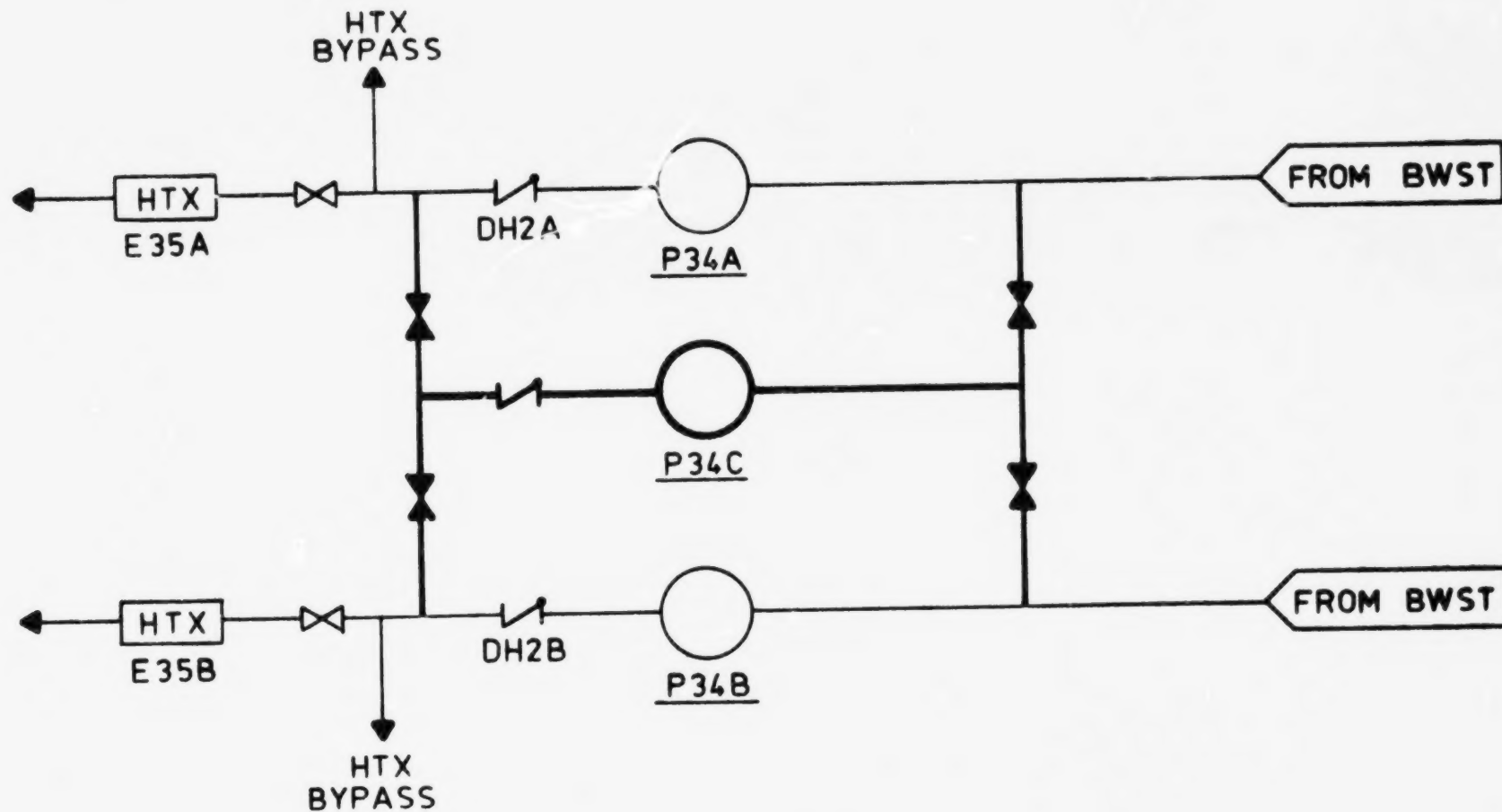
6.2 Activities After Plant Shutdown

• Tie in piping to existing DHR System	XX
• Make electrical ties	XX
• Perform hydrostatic test	XX
• Install status lights in main panel	XX
• Perform performance test	XX

TABLE 1

BILL OF MATERIAL

<u>Material</u>	<u>Quantity</u>
RHR Pump - 2500 GPM @ 400 Ft. TDH 350 HP	1
10" Sch. 40 SS Pipe	50 Ft.
8" Sch. 40 SS Pipe	50 Ft.
8" Gate Valve - Manual 600# Class Stainless Steel, OS&R type, Butt weld ends	2
10" Valves same as above	2
480V Switchgear Breakers with control switches and indicating lights	2
Annunciator - 6 window (2X3) 125VDC	1
Power Transfer Switch - 4160V, 100 Amp, 3 phase with auxiliary switch for position indication	2
Power Cable #6 AWG 5 KV	460 Ft.
Control Cable	200 Ft.
Conduit	400 Ft.
Seismic Supports	40



NEW DHR PUMP

FIGURE 1

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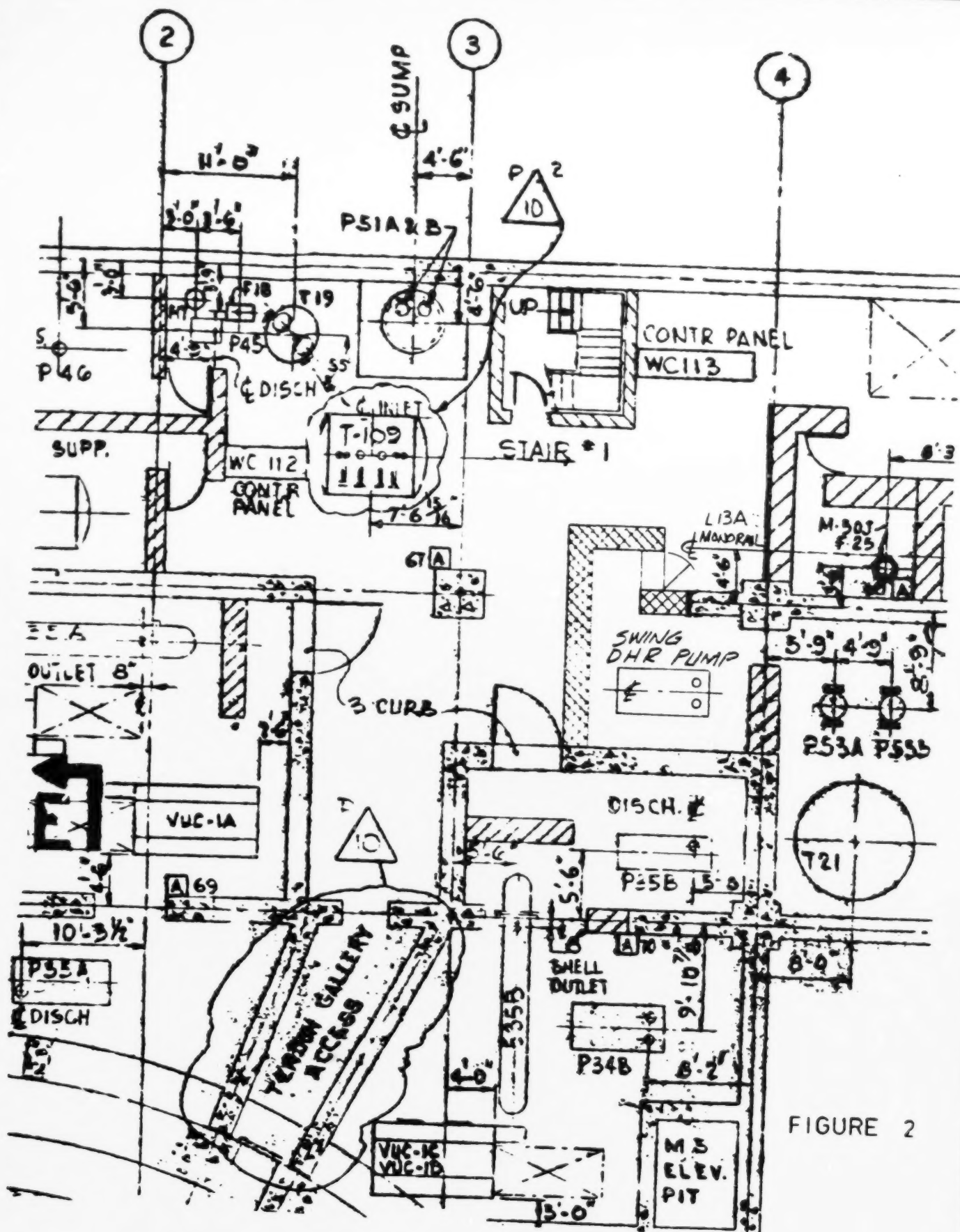


FIGURE 2

EXIST 4160V SWGR A3

EXIST 4160V SWGR A1

100

LOCAL CTL SWITCH
W/RED & GREEN LIGHTS

CTL RM STATUS
INDICATOR

100

LOCAL CTL SWITCH
W/RED & GREEN LIGHTS

CTL RM STATUS
INDICATOR

POWER TRANSFER SWITCH

CTL RM SWITCH
POSITION
INDICATOR

SINGLE LINE

350

DHR PUMP 3 MOTOR

350HP

DHR PUMP 3 MOTOR

POWER TRANSFER SWITCH

4160V SWGR
A3

4160V SWGR
A4

CTL RM
PANEL

----- CONTROL CABLE
————— POWER CABLE

BLOCK DIAGRAM

FIGURE 3

J-79

809

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ATTACHMENT D

MODIFICATION DESIGN REPORT

for

ADDITION OF REDUNDANT BWST SUPPLY VALVES

J-80

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810

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DESIGN REPORT
for
ADDITION OF REDUNDANT BWST SUPPLY VALVES

1.0 FUNCTIONS AND DESIGN REQUIREMENTS

1.1 Function and Purpose of Modification

Reduce the plants vulnerability to loss of the BWST, due to valve failure, resulting in the loss of safety injection systems.

1.2 Design Requirements and Criteria

1.2.1 Performance

All valves and piping will be designed for the same flow and pressure drop requirements as the existing system.

1.2.2 Safety/Regulatory/Availability

Safety Class 3 (Mechanical)
Safety Class 1E (Electrical)
Seismic Category I
Continuous Availability

1.2.3 Environmental

Location: Indoors
Temperature: 55-120°F

1.2.4 Installation

Valves and piping shall be accessible for inspection and maintenance.

1.2.5 Controls

New air operated valves will receive an actuation signal from the same sources as valves 1407 and 1408. Valve position shall be indicated locally and in the main control room. The new air operated valves shall be operable from a local panel and shall fail open on loss of air.

1.2.6 Testing

All new piping and valves shall be hydrostatically tested after installation. The air operated valves will require periodic testing in accordance with the ISI requirements of valves 1407 and 1408.

1.3 Design Codes

ASME BPV Code Section III, Subsection ND.
IEEE Standard 308

1.4 References

AP&L Dwg. M-232 Piping and Instrument Diagram
Decay Heat Removal System - Rev.24, dated 9/24/84.

2.0 DESIGN DESCRIPTION

2.1 Modification Configuration

This modification provides a redundant flow path for safety injection by installing a parallel valve station in the main discharge line from the BWST. A manual valve is installed in a parallel line to the existing BWST discharge, upstream of two parallel air operated valves which feed one safety injection train in parallel with existing valves CV 1407A and CV 1408B (see Figure 1). The new valves are installed in the pipe chase on the east end of the Auxiliary Building on Elevation 335.

2.2 Operation/Performance

The new manual valve is used only for maintenance and testing. It is locked in the open position during plant operation.

Each new control valve will be capable of passing full safety injection design flow.

2.3 Controls

The new air operated valves will receive a safety injection signal from the channels which feed the opposite train existing valve. That is new CVA and existing CV 1408B will be powered from the same train and CVB and CV 1407A will be powered from the other train. (See Figure 2).

3.0 COMPONENT DESIGN BASIS

3.1 Piping

Material: 304 SS Schedule 40
Pressure: 150 psig
Temperature: 100°F
Fluid: Borated Water

3.2 Valves

The new valves will be duplicates of existing valves as nearly as possible except for air operators on the new control valves. All valves will be designed and fabricated to ASME III Subsection ND requirements.

VA and B Size 14"

Material:	304 SS
Design Pressure:	150 psig
Design Temperature:	100°F
Type:	GATE
Operator:	Diaphragm: Air operated with integral Class 1E solenoid valve.

VC - Size 14"

Material:	304 SS
Design Pressure:	150 psig
Design Temperature:	100°F
Type:	Manual OS&Y Gate

3.3 Electrical

- Local Control Panel 12"x12"x8" dp Box W/2 Control Switch & Indicating Light Assemblies, terminal blocks, etc.
- 2 Status Light Assemblies for control room panel
- 650' Control Cable
- 200' Conduit

4.0 INTERFACE REQUIREMENTS

- ° 100 psig Instrument Air Supply
- ° 1E DC power for solenoid valves

5.0 SCOPE OF WORK

5.1 Installation

- Isolate and drain affected piping
- Cut lines at tie in locations
- Core bore through necessary walls
- Install piping and valves
- Install pipe supports
- Install local panel
- Install instrument air tubing
- Install cable and conduit
- Install indicating lights in main control room
- Perform hydrostatic test of piping and valves
- Perform operational check of valves and control circuits

6.0 SEQUENCE OF ACTIVITIES AND SCHEDULE

6.1 Activities Before Plant Shutdown

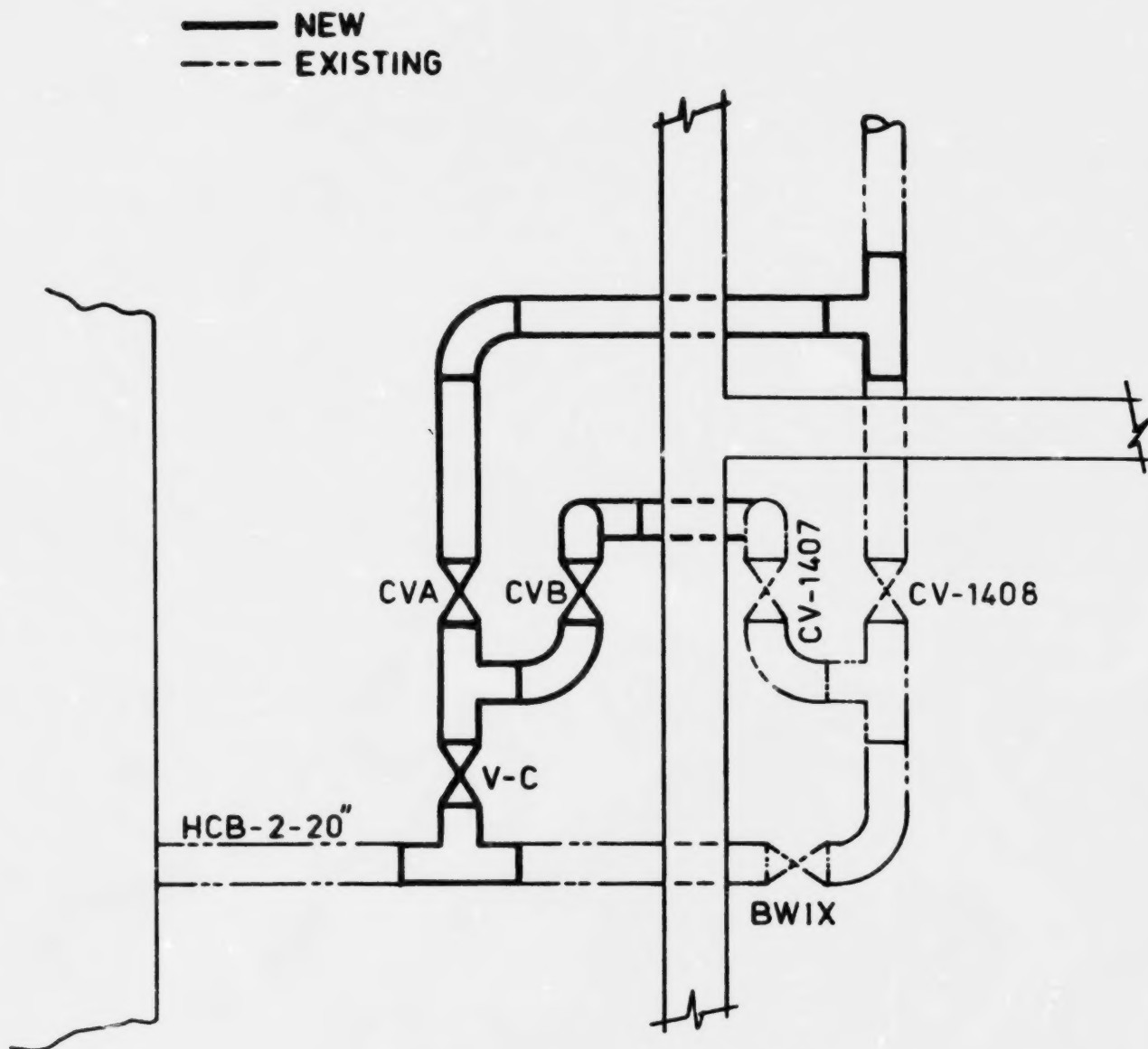
- 6.1.1 Prefabricate piping and valve segments
- 6.1.2 Partial installation of pipe supports
- 6.1.3 Install local panel
- 6.1.4 Install cable and conduit
- 6.1.5 Partially install instrument air tubing and supports
- 6.1.6 Locate and perform core bores

<u>Start</u>		<u>Shutdown</u>
6.1.1	XXXX	
6.1.2	XX	
6.1.3	XX	
6.1.4	XXXX	
6.1.5	XXXX	
6.1.6	XXXX	

6.2 Activities After Plant Shutdown

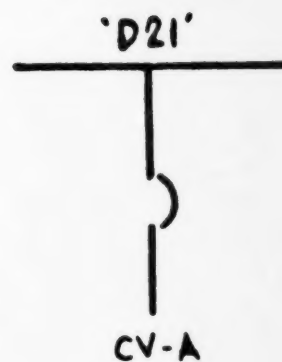
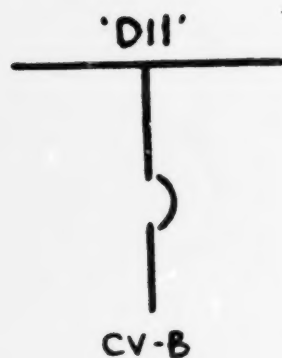
- 6.2.1 Fill refueling canal and drain the remainder of the BWSP and the affected piping.
- 6.2.2 Cut existing piping at the interface locations
- 6.2.3 Install prefabricated piping sections
- 6.2.4 Perform NDE on welds as necessary and hydrostatic test
- 6.2.5 Complete pipe support installations
- 6.2.6 Connect cable and instrument air tubing to valves VA and VB
- 6.2.7 Complete tie-in to instrument air system
- 6.2.8 Install indicating lights in the main control room
- 6.2.9 Complete wiring terminations
- 6.2.10 Perform circuit checks

	<u>Shutdown</u>		<u>Startup</u>
6.2.1		XX	
6.2.2		X	
6.2.3		XXX	
6.2.4		XX	
6.2.5			XX
6.2.6			XX
6.2.7			X
6.2.8		XX	
6.2.9		XXX	
6.2.10			XXX

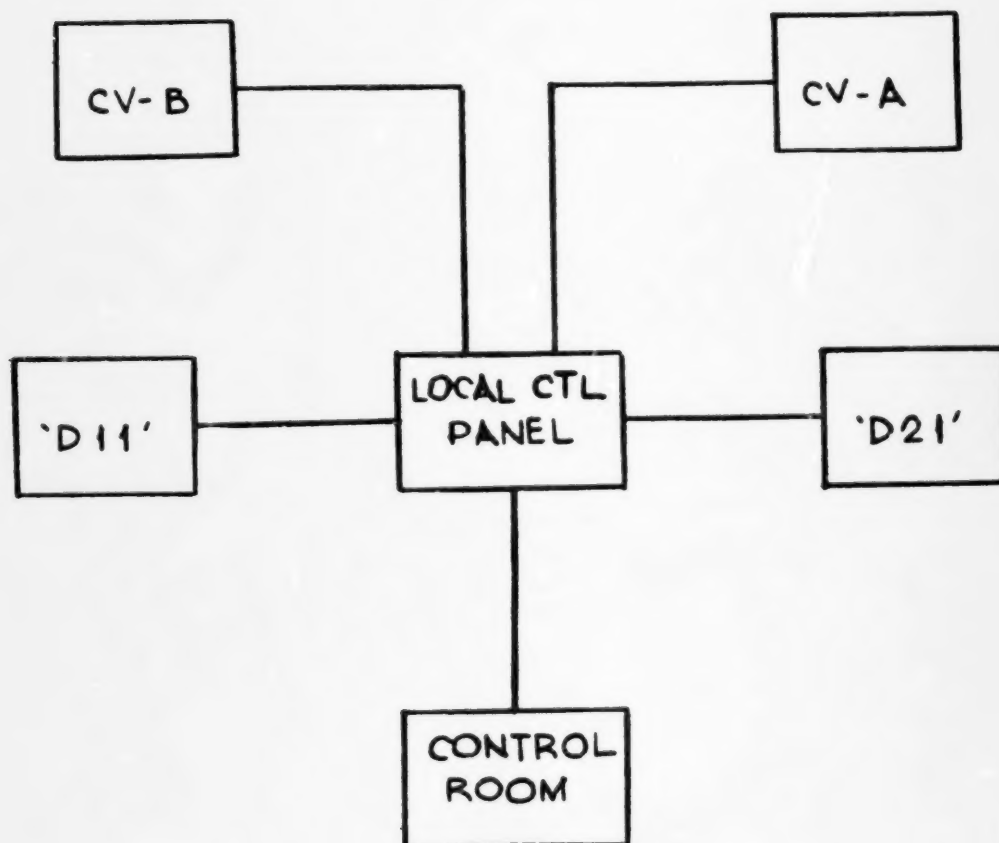


- 1-14" MANUAL GATE VALVE
- 2-14" CONTROL VALVES, AIR OPERATED W/ SOLENOIDS
- 20 FT-14" STAINLESS STEEL SCH. 40 PIPE
- 2 SUPPORTS
- 100 FT-1/2" STAINLESS STEEL TUBING

FIGURE 1



SINGLE LINE DIAGRAM



BLOCK DIAGRAM

FIGURE 2

ATTACHMENT E

MODIFICATION DESIGN REPORT

for

IMPROVED SEISMIC ANCHORAGE OF ELECTRICAL EQUIPMENT

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DESIGN REPORT
for
IMPROVED ANCHORAGE ANCHORAGE OF ELECTRICAL EQUIPMENT

1.0 FUNCTIONS AND DESIGN REQUIREMENTS

1.1 Functions and Purpose of Modification

providing seismic anchorage for the following electrical equipment:

- a) 4160 V Switchgear Cabinets (2)
- b) Control Room Panels (13)
- c) 480 V Transformer Cabinets (2)
- d) 480 V MCC's (2)

1.2 Design Requirements and Criteria

1.2.1 Safety

Anchorage shall be seismic Category I.

1.2.2 Installation

The equipment to be anchored is located in the 4160V Switchgear Room and the Control Room in the Auxiliary Building. Most of the installation will need to be done with the plant shutdown.

1.3 Design Codes

ACI 349-80

2.0 DESIGN DESCRIPTION

2.1 Modification Configuration

To add anchorage or strengthen existing anchors, steel angles will be welded to the base of electrical equipment. Holes will be drilled in the concrete floors and expansion anchors added. In cases where some anchorage is already provided, the existing anchorage will be left in place because original, embedded anchorage is inherently stronger than expansion anchors.

2.1.1 4160 V. Switchgear Cabinets

There are two switchgear assemblies in the Switchgear Room which contain the Safeguard Buses. Each assembly will be anchored along the twin sides by drilling and anchoring the existing channels to the concrete floor with expansion bolts.

2.1.2 480 V. Transformer Buses

There are two 480V transformer cabinets and two Motor Control Centers in the Switchgear Room. Each unit will be anchored by welding angles in the ends of the cabinet. The angles will be anchored to the concrete floor by expansion bolts.

2.1.3 Control Room Panels

There are thirteen console and panel cabinets in the control room. Each cabinet will be anchored to the floor by either drilling through the existing angle braces or by adding channels to the frame to use with existing embedded anchors.

ATTACHMENT F

MODIFICATION DESIGN REPORT

for

CONDENSATE STORAGE AND BORATED WATER STORAGE

TANK SUPPORTS

J-94

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3.0 COMPONENT DESIGN BASIS

Not applicable.

4.0 INTERFACE REQUIREMENTS

None

5.0 SCOPE OF WORK

5.1 Installation

5.1.1 4160 V. Switchgear

a) Drill four holes through the existing channels in the base of the frame and install four expansion anchors in each cabinet (a total of 88 bolts).

5.1.2 480 V. Transformer Cabinets and MCC's

a) Weld 4x4x1/4 angle on the ends of each cabinet and drill holes for expansion anchors.

b) Install anchors (total of 8)

5.1.3 Control Room Panels

a) Drill existing angle or weld new channel as necessary.

b) Install anchors (total of 52)

6.0 Sequence of Activities and Schedule

6.1 Activities Before Plant Shutdown
None

6.2 Activities After Plant Shutdown

All activities described in Section 5.0 will be performed after the plant is shutdown so that the equipment can be de-energized without affecting plant operation.

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--- FIGURE 1	

DESIGN REPORT
for
CONDENSATE STORAGE AND BORATED WATER STORAGE
TANK SUPPORTS

1.0 FUNCTION AND DESIGN REQUIREMENTS

1.1 Function and Purpose of Modification

Reduce plant vulnerability to a seismic event by providing additional seismic supports for the condensate storage tank (CST) and borated water storage tank (BWST) for Unit 1.

1.2 Design Requirements and Criteria

1.2.1 Safety

The supports shall be Seismic Category I.

1.2.2 Installation

The CST is located to the southwest of Unit 1 containment building on grade. The BWST is located to the northwest of Unit 1 containment building on grade.

1.3 Design Codes

AISC Specification for the Design, Fabrication and Erection of Structural Steel for Buildings.

1.4 References

Bechtel Drawing 6600-M-5, Equipment Location, Ground Floor Plan, Rev. 18, 11/21/85.

2.0 DESIGN DESCRIPTION

2.1 Modification Configuration

The CST and BWST are currently anchored by anchor bolts.

In order to increase the strength of both tanks, stiffeners will be added as shown in Figure 1. This modification applies to the CST and BWST for Unit 1-(total 2 tanks).

3.0 COMPONENT DESIGN BASIS

Not applicable

4.0 INTERFACE REQUIREMENTS

None

5.0 SCOPE OF WORK

5.1 Installation

- a) Drill bolt holes in concrete support pads
- b) Fabricate base support plates
- c) Weld support beams to base plates and to tank

5.2 Rip-out and Relocation

None

5.3 Bill of Materials

Quantities are given for a total of two tanks

Structural Steel

15' 4 x 13 wide flange	16
5' 4 x 13 wide flange	16
10 x 10 x 3/4 base plates	32
Drill holes (into concrete pads)	
13/16" diameter	128

Bolts A325

3/4 x 8" Multi Standard Kwik Bolts	128
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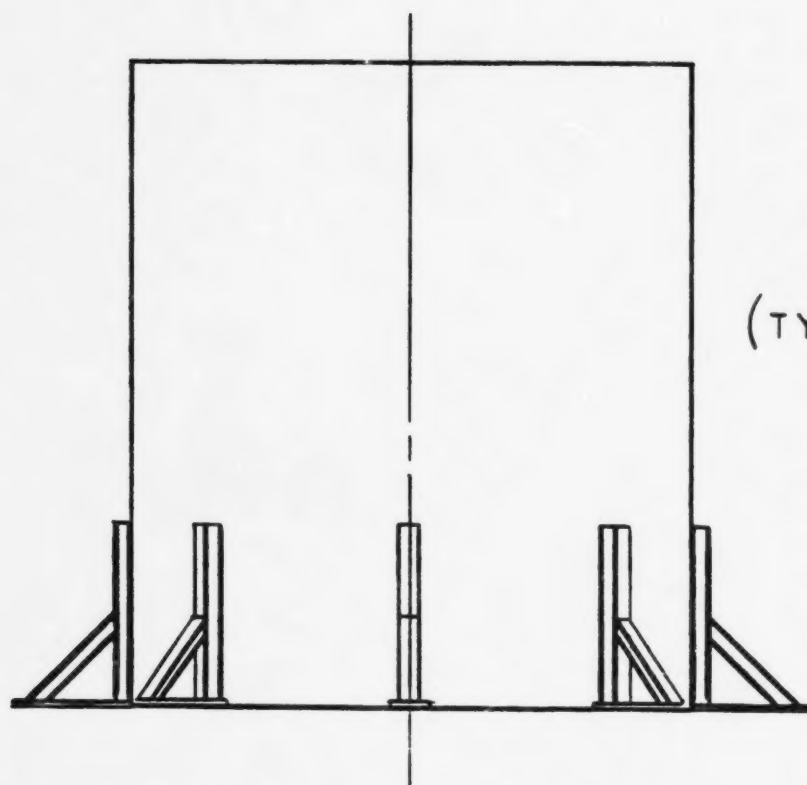
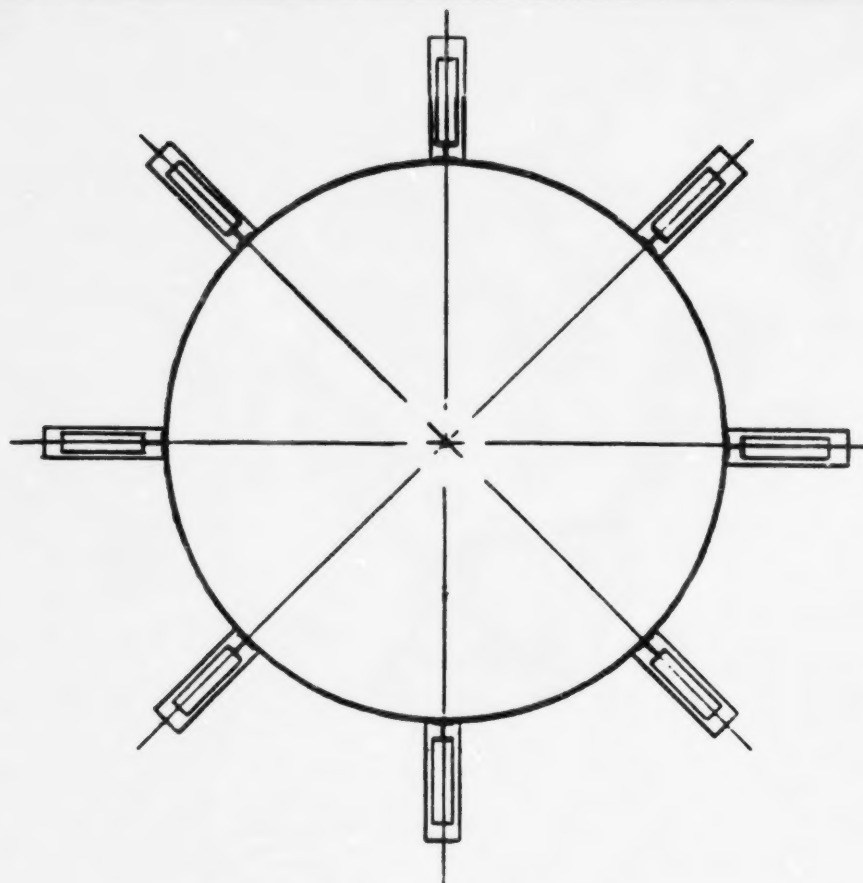
6.0 SEQUENCE OF ACTIVITIES

6.1 Activities Before Plant Shutdown

<u>Task</u>	<u>Shutdown</u>
Locate and drill anchor bolt holes	XX
Fabricate base plates	XX
Weld wide flanges to base plates and tanks	XXX

6.2 Activities After Plant Shutdown

None



(TYPICAL OF 2)

CST & RWST SEISMIC SUPPORTS

J-98-A

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APPENDIX G

MODIFICATION DESIGN REPORT

for

CABLE SPREADING ROOM

FIRE PROTECTION

SYSTEM

J-99

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DESIGN REPORT
CABLE SPREADING ROOM
FIRE PROTECTION
SYSTEM

1.0 FUNCTION AND DESIGN REQUIREMENTS

1.1 Function and Purpose of the Modification

To eliminate the vulnerability of fire damage to multiple safety systems due to a fire in the cable spreading room concurrent with failure of the existing system.

1.2 Design Requirements and Criteria

1.2.1 Performance

The modification shall assure actuation of the cable spreading room fire protection system by adding redundant deluge valves and detectors.

1.2.2 Safety/Regulatory/Availability

Non-Safety Class
Availability shall be continuous
Unavailability will result in Tech Spec limitation.

1.2.3 Environmental

Location	- Indoors
Temperature	- 50 - 100°F
Seismic	- Seismic Category I
Radiation	- Low

1.2.4 Installation

Piping and detectors shall be located in the cable spreading room.

Equipment and deluge valves shall be accessible for maintenance.

1.2.5 Control

Controls shall be provided which are designed consistent with the existing cable spreading room fire protection system.

1.2.6 Testing

Periodic inspections and tests to insure that the fire suppression system and associated alarm devices are in good operating condition.

1.3 Design Codes

The fire suppression system shall be in accordance with the National Fire Protection Association Standards.

The equipment shall be qualified and supported in accordance with IEEE-344-1975 and the plant seismic criteria.

1.4 References

• Drawings

Bechtel Dwg. No.	M-219 Piping and Instrument Diagram - Fire Water Rev. 27 dated 1/19/86
------------------	---

2.0 DESIGN DESCRIPTION

2.1 Modification Configuration

This modification will increase the reliability of the existing fire protection system actuation by adding a redundant deluge valve in the main header. The new valve will be activated by a new, separate sensing system and also will be controlled from a new local panel.

The new deluge valve and isolation valves will be installed in parallel with the existing valves (Figure 1).

There will be a total of 1000 ft. of detectowire for alarm and automatic actuation of the new valve. These will operate in conjunction with the existing detection systems.

2.2 Operation/Performance

As the presence of products of combustion in the vicinity of the trays, indicative of an incipient fire, is detected by any one of the existing smoke detectors, an alarm is sounded both locally and in the control room.

The operator has the option of opening the appropriate deluge valve from the local panel or from the control room. (Under normal circumstances, the operator will investigate the situation before taking action.)

If a fire develops, the temperature sensitive detectowire will automatically alarm and open the appropriate deluge valve. If, for any reason, the fire were to continue to develop, the area temperature detectors will alarm and open the appropriate new deluge valve.

A periodic electrical continuity test of the deluge valve solenoid/fire detector circuits will be performed from the local control panel by pushing a button and observing a light.

2.3 Controls

There will be one three-zone fire protection control panel Pyrotronic "System 3" or similar, serving the new deluge valve. The control panel will include a 24 V battery which will supply the required power to the control circuits independently of the external AC power supply.

The three-zone fire protection control panel will be located in a waterproof cabinet located outside the protected area.

The principal input and output signals to and from the fire protection control panels will be as follows:

Inputs:

- Temperature Fire Detection Signal
- Manual Pushbutton or Handle Valve Actuation (local and remote)

Outputs:

- Fire Alarm (local and remote)
- Deluge Valve Actuation Signal

3.0 COMPONENT DESIGN BASIS

The deluge fire protection system piping shall be designed and installed and tested in accordance with the NFPA Code and shall be seismically supported.

4.0 INTERFACE REQUIREMENTS

Electric power supply will be provided to the battery chargers in the local three-zone fire protection control panel.

Alarm functions and remote control functions for this fire protection system will be added to the existing fire control cabinet in the main control room.

5.0 SCOPE OF WORK

5.1 Installation

- 5.1.1 Install branch lines departing from the main header and extending through the new deluge valve and two gate valves for isolation.
- 5.1.2 Install 1000 ft. of detectowire distributed over the cable trays in the cable spreading room.
- 5.1.3 Install a three-zone fire protection control panel with local alarm and control functions. The location of this cabinet will be outside the protected area.
- 5.1.4 Provide wiring interconnections between the fire detectors, local control panel and deluge valve. Provide wiring interconnections between the local cabinet and the existing Fire Protection Control Panel in the main control room.
- 5.1.5 Provide power connections for the local panel. The interconnection wiring shall consist of the following installed hardware:
 - 600' of 2" conduit with seismic support at 10' spacings
 - 350' of control cable
 - 250' of power cable

5.2 Rip-out and Relocation

None

6.0 SEQUENCE OF ACTIVITIES AND SCHEDULE

6.1 Activities Before Plant Shutdown

6.1.1 Prefabricate piping including the deluge valve and gate valves.

6.1.2 Install headers, except interconnection with existing fire protection system.

6.1.3 Install detectowire, local control cabinet and interconnecting wiring (conduit and cable).

6.1.4 Install (conduit, cable and supports) wiring from local control cabinet to main control room, except interconnection with existing control room fire protection panel.

6.1.5 Install power feed (conduit, and cable and supports) for the local control panel.

	<u>Start</u>	<u>Shutdown</u>
6.1.1	XX	
6.1.2	XXX	
6.1.3	XXX	
6.1.4	XXX	
6.1.5	XXX	

6.2 Activities After Plant Shutdown

- 6.2.1 Isolate and drain sections of existing fire protection water main as required prior to cutting lines.
- 6.2.2 Cut and remove a small section of pipe from the existing main where it is to be connected to this system.
- 6.2.3 Complete installation of 4" headers and install pipe supports.
- 6.2.4 Perform hydrostatic test of the 4" pipe headers to the deluge valve.
- 6.2.5 Complete wiring in the main control room and interconnect with the existing control room fire protection control panel.
- 6.2.6 Perform functional system test

Shutdown

6.2.1		XXX	
6.2.2		XXX	
6.2.3		XXX	
6.2.4		XXX	
6.2.5		XXX	
6.2.6			X

Table 1

BILL OF MATERIAL

<u>Material</u>	<u>Quantity</u>	<u>Remarks</u>
4" Automatic Deluge Valves (Electrical with alarm function)	1	
4" Gate Valves OS&Y, C.I., with limit switches	2	
4" Std. wt. pipe	30 ft.	
4" Pipe supports	1	
3" Flat face B.W. flanges	12	
Detectorwire	1,000 ft.	
Local three-zone Fire Protection Control Panel, Pyrotronics System 3 or equal, including Battery Rack, Alarms, Valve Actuation and Monitoring Functions enclosed in a waterproof enclosure	1	
Control Cable	350 ft.	
Power Cable	250 ft.	
Conduit	600 ft.	

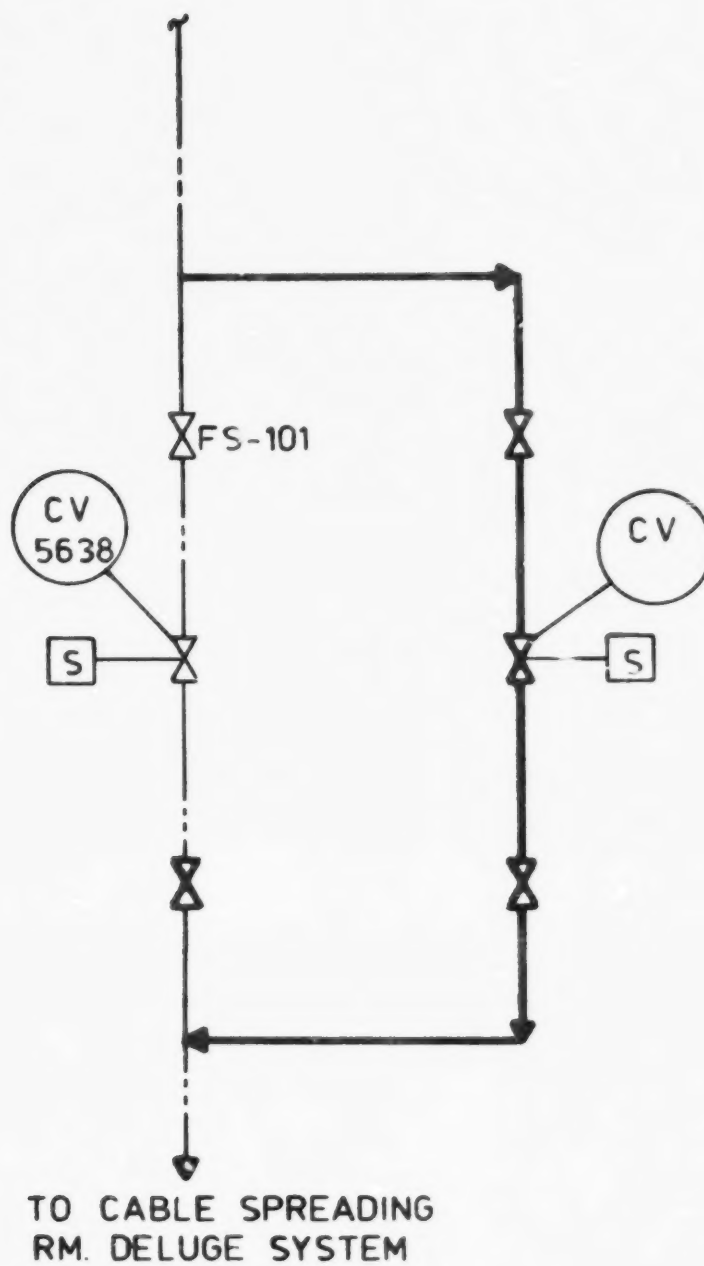
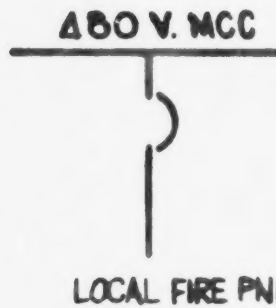
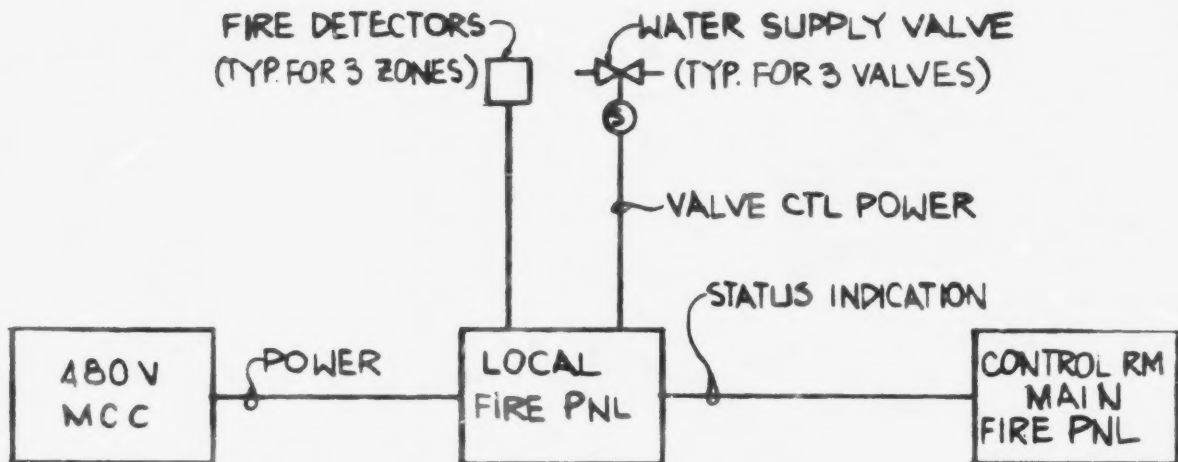


FIGURE 1



SINGLE LINE DIAGRAM



BLOCK DIAGRAM

FIGURE 2

ATTACHMENT H

MODIFICATION DESIGN REPORT

for

DIESEL GENERATOR EXHAUST SUPPORTS

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--- FIGURE 1	

DESIGN REPORT
for
DIESEL GENERATOR EXHAUST SUPPORTS

1.0 FUNCTIONS AND DESIGN REQUIREMENTS

1.1 Function and Purpose of Modification

Tie down the diesel generator exhaust piping to provide anchorage for wind (normal and tornado) and seismic loads.

1.2 Design Requirements and Criteria

1.2.1 Safety

The tie-downs shall be seismic Category I.

1.2.2 Installation

There are two diesel generators which are located on the west side of the Auxiliary Building. For both diesel generators, the exhaust piping penetrates the west wall of the Diesel Generator Room and runs vertically upwards, parallel to the exterior face of that wall. There is currently one support for the vertical portion of this piping.

1.3 Design Codes

10CFR50, Appendix A

1.4 References

Bechtel drawing:

M-7 (Rev. 9)

2.0 DESIGN DESCRIPTION

2.1 Modification Configuration

For both diesel generators, the modification will consist of adding two structural steel, strap supports per exhaust line on the unsupported vertical runs outside the building. The support will be attached to the adjacent, reinforced concrete wall with expansion bolts. See Figure 1.

3.0 COMPONENT DESIGN BASIS

Not applicable.

4.0 INTERFACE REQUIREMENTS

None.

5.0 SCOPE OF WORK

5.1 Installation

a) Install anchors

b) Install straps

5.2 Ripout and Relocation

None.

5.3 Bill of Materials

Quantities given are for two diesel generators.

Structural Steel (A36)-straps
and miscellaneous

450 lbs

Expansion Bolts - Hilti Standard Kwik Bolt,
1" diameter by 12" long

16

6.0 SEQUENCE OF ACTIVITIES AND SCHEDULE

6.1 Activities Before Plant Shutdown

Task

Shutdown

Install Anchors

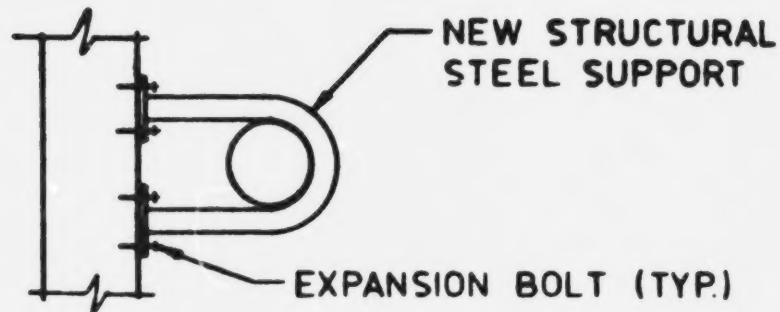
XXX

Install Straps

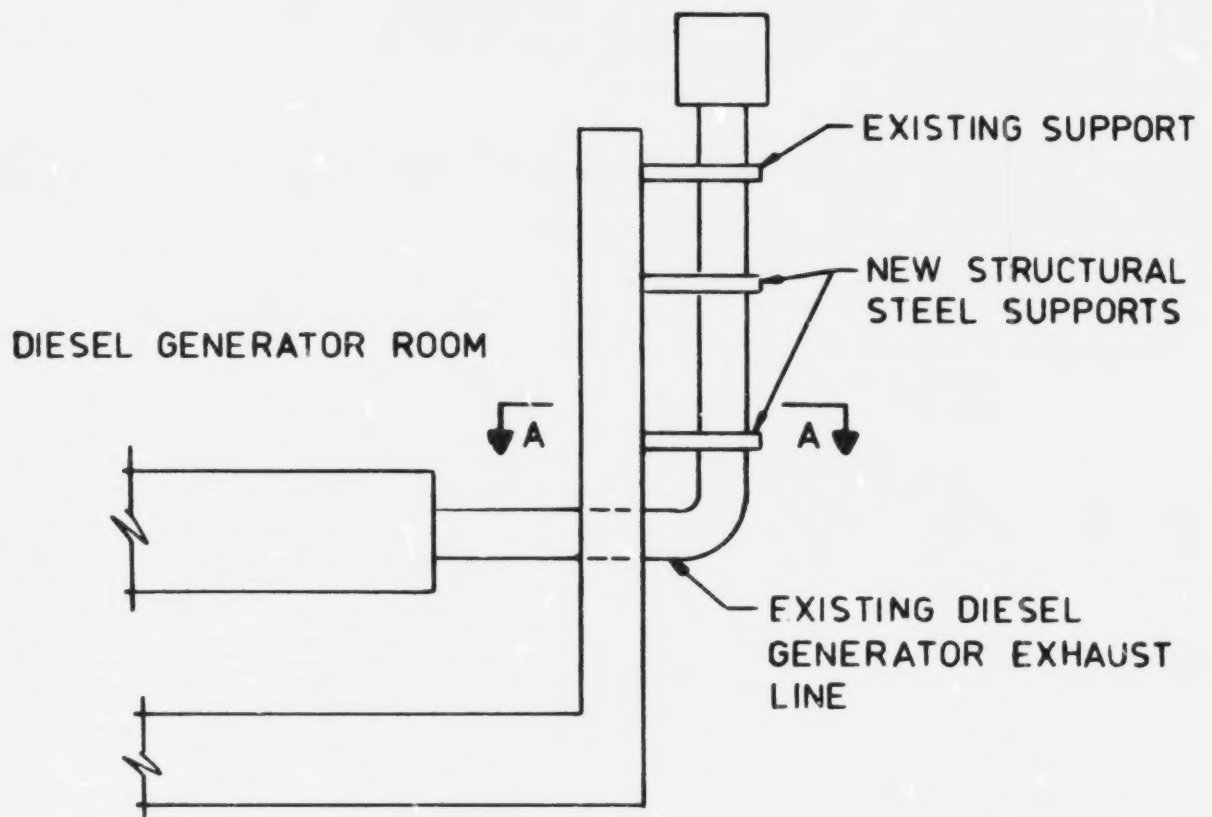
XXX

6.2 Activities After Plant Shutdown

None



SECTION 'A-A'



ELEVATION

DIESEL GENERATOR EXHAUST TIEDOWNS

FIGURE 1

ATTACHMENT I

MODIFICATION DESIGN REPORT
for the
ALTERNATIVE DECAY HEAT REMOVAL
ADD-ON SYSTEM

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DESIGN REPORT
ALTERNATIVE DECAY HEAT REMOVAL
ADD-ON SYSTEM

1.0 FUNCTION AND DESIGN REQUIREMENTS

The function of this system is to provide a new, complete, and dedicated Decay Heat Removal System to assure core cooling and reactor coolant makeup following a loss-of-coolant accident. The design requirements are as described in Reference 1 for Option 2.

2.0 MODIFICATION CONFIGURATION AND SCOPE

The discussion which follows describes Option B.2 of Reference 1 as it applies to Arkansas Nuclear One Unit 1.

2.1 Modification Configuration Description

This modification consists of a single train of feedwater and primary makeup housed within a seismic Category I structure. A schematic diagram is included in Figure 1. Both the feedwater, two 6" supply lines and the primary makeup, one 3" supply line are connected to the existing plant systems outside the containment.

The Add-on Decay Heat Removal System is completely self-contained within the ADHR tank building and ADHR structure. The ADHR tank building houses one 170,000 gal. borated water storage tank for primary makeup and one 200,000 gal. condensate storage tank for auxiliary feedwater. The ADHR tank building and ADHR structure equipment arrangements are included in Figure 2. Both structures are seismic Category I, and in addition to housing the components identified herein, they provide missile protection (turbine and tornado). The tank building is connected to the ADHR structure by a 10-foot seismic Category I tunnel. This ADHR structure houses a single train motor driven auxiliary feedwater pump, a redundant train motor driven primary makeup pump, a diesel generator complete with all auxiliaries, and the power, control and service systems necessary for operation.

The normal power supply is off-site power via the switchyard and a 34.5/4.16 kV step-down transformer located outside the ADHR structure. Power and control distribution are provided by way of the 4160V and 480V switchgears located within the ADHR structure. The diesel generator is aligned for automatic start and load transfer on the loss of off-site power. Diesel cooling is provided through a close loop service water system arranged to transfer jacket water and lube oil heat to the

atmosphere via a service water to air heat exchanger (radiator) located within the ADHR. Diesel air intake and exhaust systems are protected from missiles as are the diesel room air cooling system inlet louvers and exhausts.

The ADHR tank building and structure are located Southwest of Unit No. 1 inside the security fence as shown in Plot Plan (see Figure 3). The ADHR structure is connected to the existing auxiliary building by a seismic Category I tunnel. This tunnel contains the 3" makeup and two 6" auxiliary feedwater lines in addition to power and control conduit and cable for the motor operated isolation valves steam dump valves and instrumentation.

The tunnels, ADHR tank building to ADHR structure and ADHR structure to auxiliary building Unit No. 1, are designed for man access for maintenance and inspection. These tunnels provide missile protection for the piping and conduits.

All piping and conduit within the add-on structures, tunnels, auxiliary building and containment are seismically supported with supports located on 10'-0" centers.

The tunnel from the ADHR structure to the auxiliary building connects to the new tunnel between the new CST and the Auxiliary Building. Pipe and conduit penetrations are cut into the existing auxiliary building foundation wall and tunnel access provided by way of a locked manhole and ladder from the ADHR building grade floor.

2.2 Scope of Work

2.2.1 Basis for Outage and Non-outage Work

The scope of work is divided in two phases herein. These phases are identified as non-outage work and outage activities.

The non-outage work consists of the ADHR structures and tunnels, pipe, conduit and hanger prefabrication (all areas) and erection of the pipe, conduit, hangers, and components in the new ADHR structures and tunnels and the existing plant auxiliary building.

The outage work consists of pipe, conduit and hanger installation within the penetration areas and existing tunnel and final connections into the existing systems. This scope of work to be done during the outage is balanced between the need to avoid extending the outage and the need to keep radiation exposure to the workers to the minimum.

The ADHR structures are estimated at an 18-months construction duration. The outage activities are scheduled over 120 days or a two-outage equivalent. Utilizing two outages eliminates the need for replacement power.

2.2.2 Non-outage Activities

These activities begin with site preparation and construction of the ADHR structures and tunnels. All work is complete within the ADHR structures and tunnels, including the new off-site power supply from the switchyard. Following completion of mechanical and electrical system within the ADHR structure and tunnels, the systems are tested. The auxiliary feedwater and primary makeup lines are installed through the existing auxiliary building to the penetration areas. The new steam dump lines are installed from the existing main steam penetration through the auxiliary building roof.

Piping and conduit in the auxiliary building are installed over existing ducts, cable trays and piping to minimize interactions with existing systems and components whose failure could impact the new ADHR system. Note, existing high energy lines (if any) in the vicinity of the new ADHR system were not identified; therefore, no allowance for pipe whip restraints was included herein.

In addition to the above, prefabrication of piping conduits and supports for installation during the outage is performed at this time. Prefabrication will take place in a temporary shop adjacent to the equipment access to the Unit No. 1 containment.

2.2.3 Outage Activities

Prefabricated pipe, conduit and supports are moved into the penetration areas on Day 5 following the shutdown. This 5-day wait is included for containment purge and venting.

During the second refueling outage, the remaining pipe and conduit are installed, final connections are made and the systems are tested.

2.2.4 Rip-out and Relocation

Trailers which are located at the proposed ADHR facility location must be relocated.

3.0 SEQUENCE OF ACTIVITIES AND SCHEDULE

Non-outage

| (Start-up)

a. Site Preparation	XXX	
b. ADHR Tank Bldg. & Structure Substructure	XXXXXX	
c. ADHR Tank Bldg. & Structure Superstructure	XXXXX	
d. ADHR Structure to Auxiliary Building tunnel	XXXXXXXXXXXXX	
e. ADHR Tank Bldg. to Structure Tunnel	XXX	
f. ADHR Tank Building & Structure Mechanical Equipment Installation	XXXXXX	
g. ADHR Structure Instrumentation Installation	XXXXXXXXXXXXX	
h. ADHR Structure Elec. Equipment Installation	XXXXXXXXXXXXX	
i. ADHR Structure Grounding	XX	
j. ADHR Structure Power Supply	XXXX	
k. ADHR Structure to Auxiliary Bldg. Tunnel Piping and Conduit	XXXXXXXXXXXXX	
l. Prefabrication	XXXXXXX	XXXXXXXXXXXXX
m. Piping in Auxiliary Building		XXXXXXXXXXXXX

Outage (Containment & Pipeways)

| (Outage No. 1)

| (Outage No. 2)

n. Install Support	XXXX	XXX
o. Install Piping and Conduit	XXXX	XXXX
p. Hydro Test	851	XXXX

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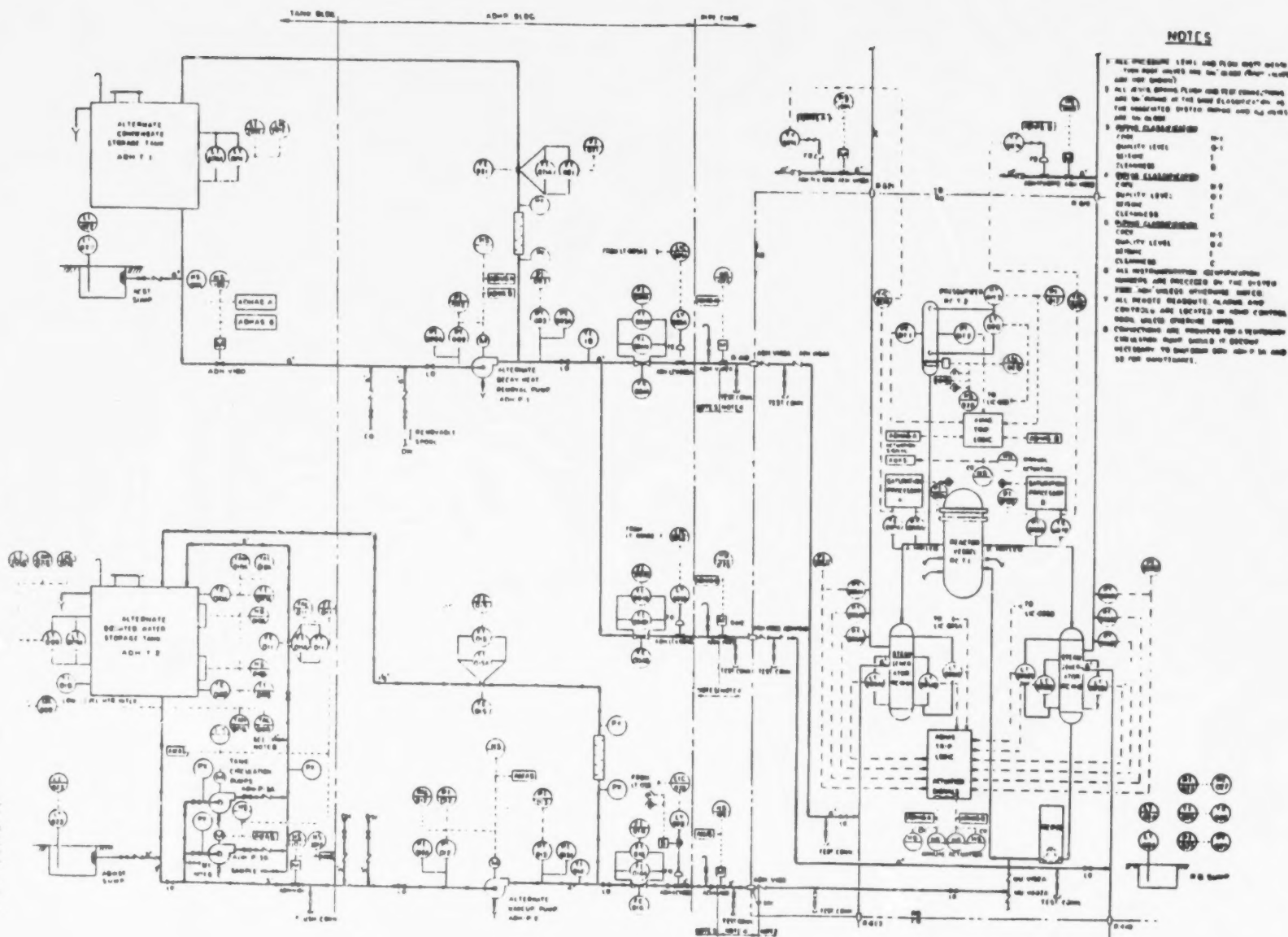


FIGURE NO. 1
SCHEMATIC DIAGRAM

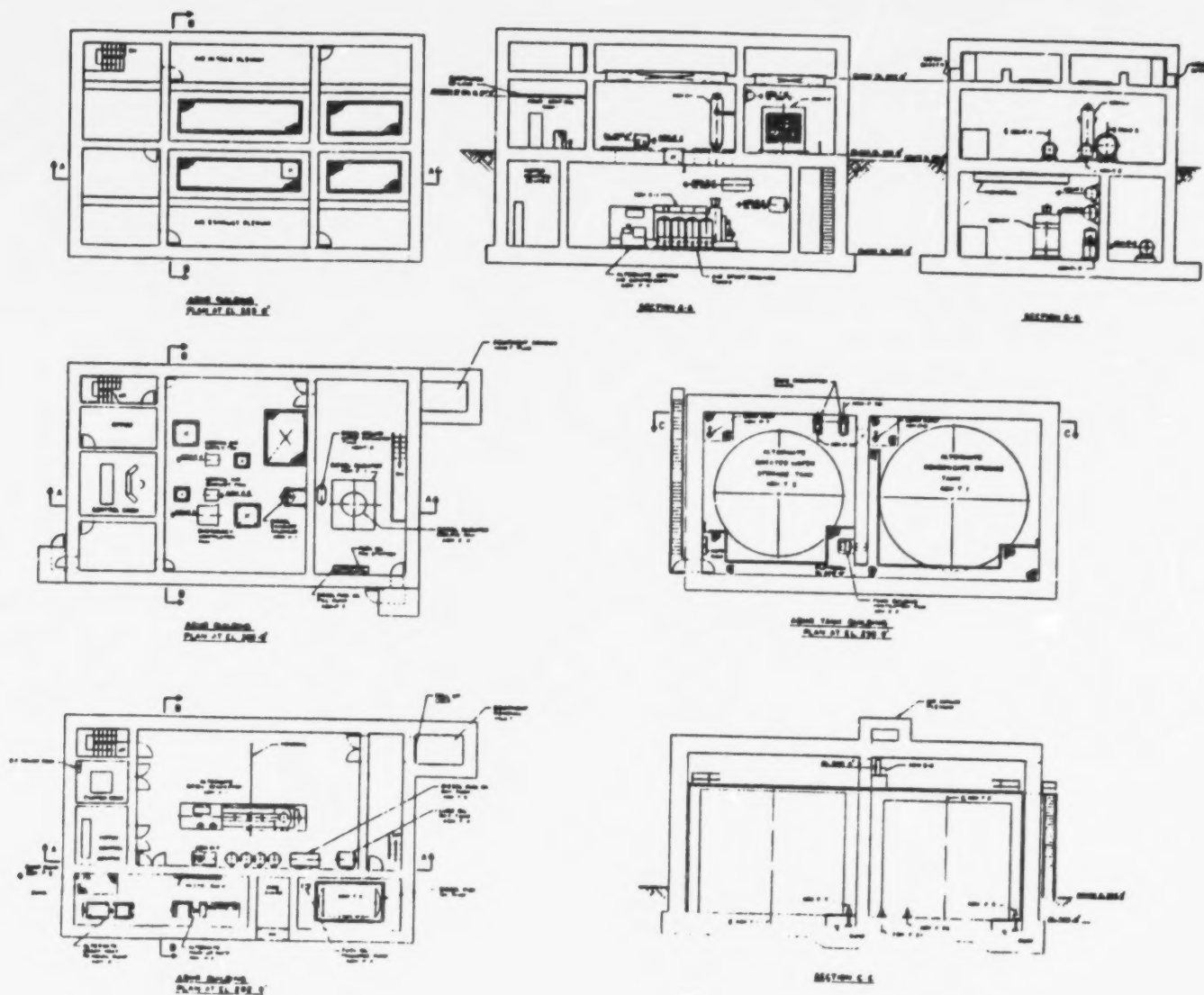
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FIGURE NO. 2
EQUIPMENT ARRANGEMENT

APPENDIX K

CONSEQUENCE EXTRAPOLATION FOR THE
ANO-1 NUCLEAR POWER PLANT

by

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1.0 INTRODUCTION

The results of the event tree, fault tree, and accident sequence analyses for ANO-1 include accident sequences, their probability per reactor-year, accident sequence cut sets, recovery factors, and proposed plant modifications which would enhance safety. In order to evaluate the consequence to the public and to perform a value-impact assessment on the proposed modifications, a method of estimating the containment response and fission product releases during meltdown accidents was developed.

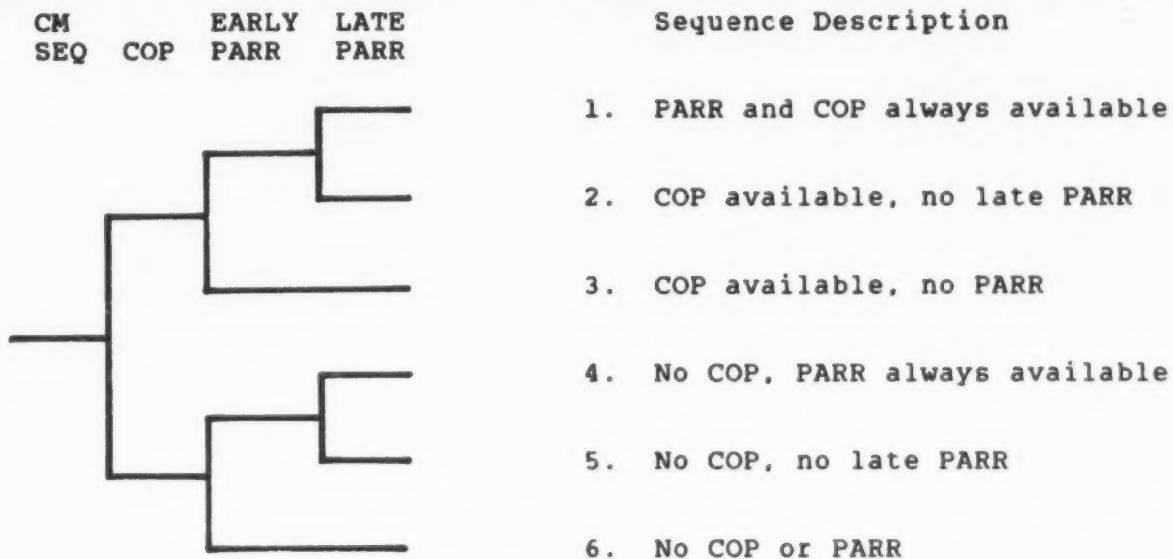
2.0 METHODOLOGY AND RESULTS

The methodology for extending the TAP A-45 PWR results to risk involved three steps: first, fault tree models of containment systems were constructed and containment system status was explicitly included in the accident sequences. This allowed the full spectrum of risk-significant containment failure modes and fission product release calculations from past PRAs were matched with the important TAP A-45 accident sequences. This matching was based upon several factors which affect containment response during an accident and is discussed below. Next, site-specific consequence calculations were made for each TAP A-45 plant. These calculations estimated the population dose out to 50 miles and the total dose that might occur given an accident. Each of the methodology steps is discussed in detail.

2.1 Containment System Modeling

The containment systems do not directly affect the probability of a core melt accident, especially when it is assumed that containment failure prior to core damage will not cause failure of emergency core cooling systems. However, containment system operability during a core melt can influence the ultimate radioactive material release and subsequent exposure, and therefore risk, to the public. Successful operation of containment sprays will lower the radioactive material release by washing out and retaining fission products inside containment. Operation of containment fans or sprays may also prevent or delay containment failure from overpressure. Any delay between initiation of core melting and containment failure provides time for deposition of airborne particles which reduces the potential offsite release.

The containment system event tree used for ANO-1 is presented in Figure K.1. The sequences delineated by this event tree were combined with the core melt sequences identified by the system event tree (refer to Appendix B, Section 5.2). The containment system event tree delineates the status of two functions: containment overpressure protection (COP) and post accident radioactivity removal (PARR). Successful COP is defined as successful containment fan and/or spray operation throughout the accident to remove decay heat from containment. PARR involves successful operation of containment sprays and is split into two



CM SEQ - Core Melt Sequence

COP - Containment Overpressure Protection

PARR - Post Accident Radioactivity Removal

K.1 Containment System Event Tree

time frames, early and late, to appropriately account for core melts which begin during the injection phase as opposed to the recirculation phase. Knowing the status of COP and PARR during a severe accident is the starting point for estimating containment failure modes and accident releases for ANO-1.

2.2 Containment Failure Modes and Fission Product Releases

Several probabilistic risk assessments have been published which include calculations of containment response during accidents, calculations of the failure modes and probabilities of containment failure based on this response, and determination of the fraction of core radionuclide inventory which is released from the containment after failure. These calculations are typically generated with computer codes such as MARCH (Reference 1), and CORRAL (Reference 2) and can easily require efforts of a man-year or more to accomplish for one plant according to the PRA Procedures Guide (Reference 3).

In order to assure a timely resolution of TAP A-45 and to accomplish this within reasonable resources it was decided to use existing, published information to estimate the containment failure modes and fission product releases for the TAP A-45 plants. The following paragraphs describe these estimates and the information upon which they were based.

Containment failure modes and fission product release results from the six plants analyzed in the Reactor Study Methodology Applications Program (RSSMAP) (Reference 5) were used. These plants and some of their characteristics are listed in Table K.1.

Table K.1

<u>Plant</u>	<u>PRA</u>	<u>Vendor</u>	<u>Containment Type</u>
Surry PWR	RSS	West	Subatmospheric
Peach Bottom BWR	RSS	GE	Mark I
Sequoyah PWR	RSSMAP	West	Ice Condenser
Oconee PWR	RSSMAP	BW	Large Dry
Calvert Cliffs PWR	RSSMAP	CE	Large Dry
Grand Gulf BWR	RSSMAP	GE	Mark III

Only a portion of the information in the RSS and RSSMAP PRAs was used without modification. Because these studies are from 2 to 10 years old, it was felt that modifications to the results to reflect newer data and analyses were warranted. These modifications include reevaluating containment failure mode probabilities based on newer estimates of containment failure pressure and utilizing updated MARCH and CORRAL code calculations performed for the RSS plants and reported in NUREG-0773 (Reference 6).

A method of selecting which of the RSS or RSSMAP plants was the best approximation for the TAP A-45 plants in terms of containment response and radioactive material release during accidents was developed. This method involves grouping the plants by containment type and then comparing the ratio of containment failure pressure (P) times containment free air volume (V) to the thermal rating of the plant (MWt). This ratio, PV/MWt, approximately relates the amount of energy the containment can absorb with the amount of energy available. It is expected that plants of the same containment type and similar values of PV/MWt would have similar containment responses for a given accident. It is recognized that the PV/MWt measure is a simplistic measure with substantial uncertainties. For example, this measure does not take into account plant-specific containment and penetration design differences which can affect the results. Based on discussions with containment response experts, however, the method we have chosen is an appropriate first order approximation.

Based on the PV/MWt ratio and containment type, it was concluded that Calvert Cliffs was an approximation to ANO-1 in terms of containment response to accidents. The results which were used for ANO-1 are given in Tables K.2 and K.3. Table K.2 presents the containment failure mode probabilities for each type of core melt accident and their associated fission product release category. Table K.3 presents the fission product release fraction (fraction of total core inventory which is released) for each of the release categories. Note that the release fractions for a given accident type may vary from plant to plant due to plant-specific design features.

2.3 Accident Sequence Consequences

Using the estimates of the fission product releases described in the last section, the potential consequences that could result from an accident occurring at ANO-1 were calculated. The calculations were performed using the CRAC code (Reference 7). The population distribution around the ANO-1 facility and the local weather patterns were used as input to the CRAC analysis.

The primary CRAC code result used in the ANO-1 analysis is the integrated dose in man-rem received by the population around the plant after an accident integrated out to a distance of fifty miles.

Three different sets of CRAC results were calculated. The first calculation uses the release fractions described in the last section and is called the "upper bound" calculation. For the second calculation, all of the release fractions except for the noble gases were uniformly reduced by a factor of seventy percent (0.3 times the upper bound). This is called the "baseline" calculation. For a final "lower bound" calculation,

Table K.2

**Modified Containment Failure Mode Probabilities
and Release Category Assignments**

<u>Accident Type</u>	<u>Containment Failure Mode Probability</u>	<u>Release Category</u>
1. Core melts with contain- ment sprays and heat removal always available.	$\alpha = 0.0001$ $\epsilon = 0.25$ $\gamma + \delta_e = 0.014$ $\delta_l = 0.18$ $\beta = 0.002$	1 7 3 5 5
2. Early core melts with containment heat removal always available. Sprays succeed initially, fail in recirculation.	$\alpha = 0.0001$ $\epsilon = 0.25$ $\gamma + \delta_e = 0.014$ $\delta_l = 0.18$ $\beta = 0.002$	1 6 3 4 5
3. Core melts with contain- ment heat removal available, no sprays at time of core melt.	$\alpha = 0.0001$ $\epsilon = 0.25$ $\gamma + \delta_e = 0.014$ $\delta_l = 0.18$ $\beta = 0.002$	1 6 2 3 4
4. Early core melts with no containment heat removal, sprays always available.	$\alpha = 0.0001$ $\epsilon = 0.25$ $\gamma + \delta_e = 0.014$ $\delta_l = 0.18$ $\beta = 0.002$	1 7 3 4 5
5. Late core melts with no containment heat removal, sprays succeed early, fail late.	$\alpha = 0.0001$ $\epsilon = 0.25$ $\gamma + \delta_e = 0.014$ $\delta_l = 0.18$ $\beta = 0.002$	1 6 3 4 5
6. Core melts with no containment heat removal, sprays succeed prior to melt, fail at melt.	$\alpha = 0.0001$ $\epsilon = 0.25$ $\gamma + \delta_e = 0.014$ $\delta_l = 0.18$ $\beta = 0.002$	1 7 2 3 4
7. core melts with no containment heat removal, or sprays.	$\alpha = 0.0001$ $\epsilon = 0.25$ $\gamma + \delta_e = 0.014$ $\delta_l = 0.18$ $\beta = 0.002$	1 6 2 3 4

Containment Failure Modes

α In-vessel steam explosion	β Containment leakage
γ Hydrogen burn overpressure	ϵ Base mat melt through
δ_e Ex-vessel steam spike	δ_l Steam and noncondensable gas overpressure

Table K.3 ANO-1 Source Term

Release Category	*Fission Product Release Fraction						
	Xe	I	Cs	Te	Ba-Sr	Ru	La
1	1E-0	7E-1	4E-1	4E-1	5E-2	4E-1	3E-3
2	1E-0	5E-1	8E-1	7E-1	9E-2	6E-2	9E-3
3	1E-0	4E-2	2E-1	6E-1	9E-3	4E-2	7E-3
4	1E-0	4E-2	4E-2	3E-2	5E-3	3E-3	4E-4
5	1E-0	3E-2	9E-3	5E-3	1E-3	6E-4	7E-5
6	1E-0	8E-4	8E-4	1E-3	9E-5	7E-5	1E-5
7	3E-6	2E-5	1E-5	2E-5	1E-6	1E-6	2E-7

* Based on RSSMAP and RSS Calculations

all of the release fractions except noble gases were uniformly reduced by ninety percent (0.1 times the upper bound). These additional calculations were performed to gauge the sensitivity of the potential consequences to variations in the source term. This approach was deemed appropriate in light of the ongoing discussions on reactor accident source terms. The results of the three CRAC code calculations for ANO-1 are given in Table K.4

It can be noted from Table K.4 that the 50 mile population doses are relatively insensitive to variations in the source term. In all cases there is less than a factor of three difference between the lower bound and the upper bound values. This insensitivity occurs because there are competing effects with the early and long-term doses received by the public. The early exposure results from the initial plume passage and is nonlinear because of the threshold and dose rate effects. However, the early exposure is a function of the source term and will decline with it. The long-term exposure results from food and crop contamination and from the assumption that the public will be allowed to reoccupy decontaminated land at an exposure rate of not more than 25 rem in 30 years. The long-term dose is therefore not as strongly coupled to the magnitude of the source term. In fact, as the source term is reduced, the CRAC code will assume that less land is interdicted and that more land will be decontaminated and reclaimed for public use. This means that the number of people who may potentially receive 25 rem in 30 years by living on decontaminated land may, in fact, increase as the source term is reduced. This factor competes with the reduction in early dose. This effect is magnified by looking at the 50 mile population dose because the decontamination radius is often greater than 50 miles. Because there is so much more land area beyond the decontamination distance in which varying amounts of exposure can occur, the total dose more closely follows changes in the source term.

2.4 Use of Consequence Results in Value-Impact Analyses

The measure of risk which is used in value-impact analysis for ANO-1 is the present worth of the expectation value of radiation dose to the public for the remaining lifetime of the plant. The conditional population doses described in this chapter are used in this calculation. The expectation value of population dose is calculated by summing all the accident sequence probabilities times their containment failure mode probabilities times the appropriate fifty mile dose associated with the containment failure modes. Refer to Appendix L and Section 9 for detailed discussions on the value-impact analysis.

Table K.4 ANO-1 Conditional Consequences

Release Category	Lower Bound Source Term		Baseline Source Term		Upper Bound Source Term	
	<u>Population Dose</u> Within 50 Mi.	<u>Total</u>	<u>Population Dose</u> Within 50 Mi.	<u>Total</u>	<u>Population Dose</u> Within 50 Mi.	<u>Total</u>
1	2.9E+5	2.1E+6	4.7E+5	4.2E+6	7.9E+5	1.2E+7
2	3.5E+5	3.0E+6	4.9E+5	5.9E+6	7.5E+5	1.4E+7
3	2.3E+5	1.4E+6	3.5E+5	2.6E+6	5.9E+5	5.0E+6
4	1.4E+5	4.6E+5	2.1E+5	1.0E+6	3.2E+5	2.2E+6
5	6.5E+4	1.5E+5	1.2E+5	3.5E+5	2.0E+5	8.5E+5
6	1.0E+4	2.7E+4	2.3E+4	5.5E+5	5.4E+4	1.4E+5
7	2.1E+3	1.2E+4	2.4E+3	1.3E+4	6.0E+3	1.4E+4

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APPENDIX L

METHODOLOGY FOR THE VALUE-IMPACT ANALYSIS
FOR THE ANO-1 POWER PLANT STUDY

by

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1.0 INTRODUCTION

One of the principal objectives of the A-45 program (Reference 1) is to produce a Regulatory Analysis which can serve as a basis for decision-making concerning the need, if any, for improvement of the decay heat removal (DHR) function in existing light water reactors (LWRs). An important part of the Regulatory Analysis is the preparation of "Value-Impact" (or "Benefit-Cost") analyses to determine whether or not proposed modifications are likely to be cost effective.

The NRC has specified certain features which it wishes to see included in Value-Impact analyses and has specified some of the parameter values that should be used (Reference 2). In addition, NRC has sponsored the development of a "Handbook" which sets out methods which should be adopted for these analyses (Reference 3). The Handbook identified three levels of detail (Limited, Intermediate, and Major Effort) at which these analyses could be conducted. The NRC requirements, as set out in Reference 2, are at the Limited Effort level of the Handbook recommendations.

However, both the NRC requirements and the Handbook recommendations leave a number of areas in which the value-impact analyst is left with some freedom of choice. In this Appendix specific recommendations are made as to the methods to be used for TAP A-45.

Most of the recommendations made in this Appendix are at the Limited Effort level of the Handbook but, in order to provide a more adequate treatment of the uncertainties, some extensions to a higher level have been made. Further, it is noted that these recommendations are based on a study of value-impact analysis which has been in progress at UCLA since 1983. Some of the generic results of this study have been reported in References 4, 5, and 6.

This Appendix also provides a description of the formats used to present the results of the value-impact analyses.

2.0 SPECIFIC NRC REQUIREMENTS FOR VALUE-IMPACT ANALYSES

2.1 The Principal NRC Requirements

The basis of the NRC approach to Value-Impact analysis is that the "Value" term should be based primarily on population dose out to a radius of 50 miles from the site, as estimated by the CRAC-2 Code (Reference 7). NRC requires that a value of \$1000 per person rem should be attributed to the potential reduction in dose arising from a modification. Value and impact are to be estimated on the basis that the effects of inflation should be disregarded but, with certain exceptions, the reduced worth of payments made or incurred at some future date should be

taken into account. For this purpose the effects of using discount rates of 5 percent and 10 percent per annum should be examined. The present worths of quantities expressible in dollars are estimated for the remaining plant lifetime.

Due to the controversy concerning the treatment of the costs arising from material damage to a nuclear power plant (NPP) resulting from a nuclear accident, the NRC requires that these should be treated as "negative impacts", rather than "positive values", and that they should be shown separately in the analysis. The avertable cost of decontamination and clean up of the NPP and site is also to be treated in this way, except for the dose which would be incurred by personnel engaged in such work; this is to be treated as a "positive value." However, occupational doses incurred in the execution of a modification are to be treated as "negative values;" changes in occupational dose during subsequent operation are also to be treated as a change in the value term.

In the absence of better data, "rule of thumb" figures for cost of clean up and for the associated personnel dose have been proposed by NRC (Reference 2). These are \$1.2 billion and 40,000 man rem respectively. A more recent study by Argonne National Laboratory (Reference 8) than that referred to in the PNL Handbook provides data on replacement power costs which are acceptable to NRC (Reference 9).

NRC has also emphasized the need to take account of features peculiar to specific sites, such as additional liquid pathways to humans, and for a careful examination of the uncertainties in the analysis.

2.2 Implementation of NRC Requirements

In the development of the value impact analyses for TAP A-45 the principle has been adopted that in general the analyses should meet the specific requirements stated by NRC in Reference 2 and the detailed procedures set out in the following sections are based on this principle. They are also consistent with the NRC direction that the methods proposed in the Handbook (Reference 3) should be adopted where appropriate. However, as indicated above, to deal adequately with the wide range of uncertainties associated with this type of work requires going beyond a "Limited" analysis and also the development of approaches and methods which are not contained in the Handbook. These departures from the Handbook, have led to recommendations to present the results in a style which differs in some respects, from the examples given in the Handbook.

Table L.1 provides a summary of the required division into "Impacts" and "Values".

Table L.1

Categorization of Attributes into "Impacts" and "Values"
for the "Limited Effort" Value-Impact Analysis for ANO-1

IMPACTS	VALUES
<u>POSITIVE</u>	<u>POSITIVE</u>
Engineering and installation costs of the modifications.	Averted offsite dose
Operations and maintenance costs resulting from the modification	
Replacement power costs during Installation	Averted on site dose due to cleanup and to additional occupational exposure on second unit
Other costs to utility arising from the modification	
NRC costs arising from modification*	
<u>NEGATIVE</u>	<u>NEGATIVE</u>
Averted on site costs (Replacement power, loss of investment, cleanup cost	Additional radiation exposures measured in carrying out modification
<u>POSITIVE OR NEGATIVE</u>	<u>POSITIVE OR NEGATIVE</u>
Changes in plant availability due to Modification (change in outage times for maintenance and in service inspection; changes in outage time due to technical specification)	Change in occupational dose due to modification

*These costs are not evaluated in the Technical Contractor's report.

3.0 METHODOLOGY EMPLOYED FOR VALUE IMPACT ANALYSIS IN TAP A-45

3.1 Subdivision of the Analysis

The work of performing the analysis is subdivided into five parts as follows:

- a) Preparation of the Impact Analysis
- b) Preparation of the Value Analysis
- c) Examination of the uncertainties in the impact and value analyses
- d) Combination of the impact and value analyses to provide a measure of the cost effectiveness of the various alternatives investigated, primarily on the basis of "Best Estimate" values
- e) Examination of the effect of the uncertainties on the conclusion reached about the cost effectiveness of the various alternatives using "Best Estimate" data.

The methodology proposed for each of these parts of the value-impact analysis is described below.

3.2 Methodology for the Impact Analysis

3.2.1 The Content of the Impact Analysis

The principal part of the Impact Analysis is the estimation of the engineering costs of executing each of the proposed Alternatives. This is essentially a deterministic process, using methods and data commonly employed in Case Engineering, which is not subject to major uncertainties. However, as in any construction work, unforeseen difficulties may arise, leading to additional cost.

The NRC requirement to treat "avertable on-site costs," as "negative impacts," rather than "positive values," necessitates an extension of the conventional Impact Analysis to cover these items, which are essentially of a probabilistic nature. The NRC also requires that the occupational dose incurred during the installation phase of an Alternative, and any changes in occupational dose during service be treated as "negative values," rather than as impacts.

Another potentially important impact term, in the case of Alternatives which increase the redundancy or diversity of the systems available to perform the DHR function, is the reduction in the outage time that could be incurred in order to comply with the Technical Specifications for the plant in the event of breakdown of equipment in the DHR systems.

3.2.2 Estimation of Engineering Costs

As indicated above, the methods employed for this part of the Impact Analysis are well established and are in constant use in the chemical and other industries; a step towards codifying their application to nuclear power plants (NPP) has recently been taken by the issuance of a "Handbook for Cost Estimating" (Reference 10). A major contribution to the Handbook was made by the Architect-Engineer (AE) who prepared the engineering cost estimates in this program. Thus, as would be expected, the methods and data employed are consistent with those in the Handbook.

In order to reduce the uncertainties in the estimates, the following steps have been taken:

- a) Information about local wage rates and the productivity of local labor, relative to the national average values, has been obtained.
- b) Visits to the specific plant have been made to confirm impressions gained from the plant drawings. Potential difficulties such as radiation hazards and congestion of working spaces have been discussed with the utility management.
- c) The same team of design and cost engineers, experienced in detailed estimating of NPP modifications, has been used to estimate the cost of each of the Alternatives.

The estimates of engineering costs include estimates of the radiation doses which would be incurred during the installation phase and the changes in occupational doses which would be expected during subsequent operation. A monetary value of \$1000 is attributed to each additional person-rem which it is expected to be incurred during the installation phase; changes in occupational dose during subsequent operations are monetarized in the same way. Because these changes are relatively certain, as compared with the statistical nature of doses that would be incurred in the event of a reactor accident, they are not discounted. These sums are transferred to the Value Analysis and treated as described in Section 3.3.4 below.

The estimates of engineering costs also include estimates of any plant outage time required for the installation phase of each Alternative. Where it is expected that the required outage time would be greater than that available during a scheduled outage for maintenance or refueling (or a series of such outages), the replacement power cost is estimated using the best available data, which are contained in the recent ANL report (Reference 8) mentioned above. The ANL report provides cost data for each of the seasons, up to Summer 1986. However, because the time at which any Alternative would be put into

effect cannot be forecast at the moment, an average value for the years 1985/1986 has been used. In the case of the ANO-1 NPP the estimated cost of replacement power is $\$3.59 \times 10^5$ per day.

Possible changes in the materials and equipment costs associated with maintenance and in-service inspection which would arise from each of the proposed Alternatives have also been estimated by the AE. There could also be changes in the outage times required for maintenance and in-service inspection. However, as in the case of possible effects on outage time arising from increased redundancy or diversity of DHR systems (see Section 3.2.1 above), quantification of these times requires additional analysis of system reliabilities. These items, together with possible outages arising from additional spurious scrams or enforced shutdowns, have been excluded from the engineering cost estimates but some comments have been included elsewhere (see Section 3.2.5 below).

3.2.3 Estimation of Other Positive Impacts

In addition to the positive Impacts identified above, there are other quantifiable costs associated with each of the proposed Alternatives; some of these would have to be borne by the utility, others would have to be met by NRC.

The principal additional costs to be met by the utility would be:

- a) Changes in labor costs associated with changes in maintenance and in-service inspection requirements.
- b) Costs arising from negotiation with NRC concerning proposed Alternatives.
- c) Costs involved in preparation of revised operating procedures and Technical Specifications and in operator training.
- d) Cost changes in subsequent operation due to changes in reporting procedures required by NRC (e.g., additional reports on in-service inspections).

It was only possible to prepare an approximate estimate of these costs. Because NRC costs, for development, implementation and operation, due to the proposed Alternatives, can only be established on a generic program basis, it was agreed with the NRC Project Manager for TAP A-45 that these should be included in the overall Regulatory Analysis, rather than in these technical reports for the individual NPP.

3.2.4 Estimation of "Negative Impacts" based on avertable onsite costs.

As indicated in Section 2.1, above, NRC prefers to have most of the avertable on site costs treated as "Negative Impacts" rather than positive values; the avertable on site costs usually included in Value-Impact analyses are as follows:

- a) Replacement power costs (i.e., the increase in the cost of generation due to the difference in the cost of nuclear fuel and that of the fuel used in lieu of nuclear, plus any difference in other operating costs).
- b) Loss of the remaining value of the utility's investment in an NPP which is not repairable, or cost of repairs and refurbishing.
- c) Cost of clean up of plant and site.
- d) Radiation exposures resulting from an accident, including cleanup operations.

All of these, except (d), are treated as "Negative Impacts".

There are several other possible causes of substantial losses, which might or might not be borne by the electric power industry: these are discussed in Reference 11.

Since the methods proposed for deriving quantitative estimates for each of Items a) through c) above differ from those described in the PNL Handbook (Reference 3) they are described in detail below.

a) Replacement power costs. It is visualized that it would require 10 years for a utility to replace an NPP in which severe core damage, or core melt, had occurred. After that time it is assumed that power would be provided by a new plant (the capital cost of which would be included in the utility's rate base in the usual way) that at the time would provide the lowest achievable generation cost (i.e., fuel cost + capital cost). Thus the replacement power at any given time is the present worth, at that time, of the differential fuel and operating costs measured in replacing the NPP output, assuming 65 percent capacity factor. At a discount rate of 5 percent, the present worth of 10 years' replacement cost is $7.72 R$, where R is the cost for one year (see Table C.2 of Reference 3).

The annual expectation of loss due to potential replacement power costs is $7.72 R p_m$, where p_m is the probability of core melt per reactor year. The event can occur at any time during the remaining life of the plant, N years. In a rigorous analysis there should be a correction for the fact that, if core melt were to occur in the last 10 years of life, the

potential loss would be reduced. However, the correction is small and has been omitted here. Thus the present worth in 1985/1986 of the potential loss is the discounted value of N payments of 7.72 Rp_m, at a discount rate of 5 percent.

The ANO unit 1 has a remaining life of 22 years based on a license expiration date of 2008. Taking this value of 22 years for ANO-1, the present worth of the maximum potential loss is given by Reference 3):

$$PW = 7.72 \text{ Rp}_m \left[\frac{(1+r)^N - 1}{r(1+r)^N} \right]$$

where:

r = discount rate = 5%

N = number of years over which loss may occur = 22

Thus for ANO-1 this value is 101.6 Rp_m.

The best available basis for estimating R is provided by the data mentioned in Section 1 above. These are forecasts up to Summer 1986; in the longer term two factors operating in opposite directions could change these:

i) It is generally expected that in the long term the price of fossil fuels will escalate more rapidly than the general inflation rate, whereas nuclear fuel prices will escalate at the general inflation rate.

ii) The proportion of low efficiency and high fuel cost plants on most systems will diminish in the longer term as old plants are retired and are replaced by more efficient plants and as oil-fired plants are converted to coal.

It is assumed, therefore, that the best available estimate can be derived from the average of the last 4 periods given in Table S.1 of Reference 8. However, the estimate for each period has to be corrected for capacity factor since the tabulated data are based on estimated seasonal capacity factors, with no allowance for planned outages. The average value of 65 percent allows for these outages. On this basis, the annual costs of replacement power (R) for ANO-1 would be \$118 x 10⁶.

Thus the maximum avertable replacement power costs, (i.e., if p_m were reduced to zero) would be:

$$\text{ANO-1} \quad \$11.9 \times 10^9 \text{ p}_m$$

Reduction of p_m by an amount Δp_m would give a pro rata saving in replacement power costs.

b) Loss of remaining value of utility's investment. Since all costs and values in the analysis are being estimated in current US dollars, it is considered that the most appropriate method for evaluating the utility's investment at the time of an accident leading to a "write off" is to assume that its original capital cost would have been similar to those of an NPP ordered in 1985/1986, approximately \$1500 per KW(e), and that it depreciates at a uniform rate over a 40 year lifetime.

Thus the estimated value of ANO-1 (836 MW_e) in 1985/1986 would be $836 \times 10^3 \times 1.5 \times 10^3 \times 22/40$, or $\$6.90 \times 10^8$.

The expectation of loss in any one year is $V_C p_m$, where V_C is the depreciated value in that year. Thus, the present worth of the maximum expected loss is the sum of the discounted expectations over the remaining life. The maximum avertable costs due to loss of investment value are:

$$\text{ANO-1 } \$3.05 \times 10^9 p_m$$

As in (a), above, reduction of p_m by an amount Δp_m would give a pro rata saving in the potential loss of investment value.

The total costs are not greatly different from those which would be obtained using the alternative approach (adopted in Reference 10) that the loss of capital invested can be represented by the total discounted value of replacement power for the remaining life of the plant at the time of the accident; the latter method yields avertable losses which are larger. However, the method used here has the advantage that it shows separately the item which would be the most likely to be passed on directly to the consumer, i.e., the replacement power costs for the first 10 years after an accident.

(c) Cost of plant and site cleanup. In the absence of any better data, the NRC "rule of thumb" figure of $\$1.2 \times 10^9$ is used. It is assumed that this represents the present worth, at the time of the accident, of expenditures that would be spread over several years.

The present worth in 1985/1986 is found in the same way as for the replacement power cost. Thus, at a 5 percent discount rate, the maximum avertable costs for cleanup are:

$$\text{ANO-1 } 1.2 \times 10^9 \times 13.2 p_m \text{ or } \$15.8 \times 10^9 p_m$$

Since the accident sequence analysis is based on core melt no attempt has been made to estimate the cost of repairing and refurbishing an NPP after a less severe accident.

(d) Total avertable on site costs. Summing the three components estimated above, the maximum total avertable on site costs are:

$$\text{ANO-1} \quad \$3.08 \times 10^{10} \text{ p}_m$$

To obtain the savings for a reduction of Δp_m in p_m , these amounts are multiplied by $\Delta p_m / p_m$, i.e., $3.08 \times 10^{10} \Delta p_m$.

3.2.5 Possible Negative Impact Due to Improvement of Availability

As indicated in Sections 3.2.1 and 3.2.2 above, a major modification to improve the DHR function may affect availability favorably by reducing the frequency of forced outages, due to an improved capability for meeting Technical Specifications requirement when repairs to DHR related plant items are in progress. However, adverse effects on availability are also possible due to an increase in the frequency of spurious reactor trips or an increase in the time required for maintenance.

An indication of the extent of the "Negative Impact" is provided by the following illustrative example: It is supposed that a modification avoids the need for a forced shutdown to meet the Technical Specification once in three years and that on each occasion an outage of 3 days is avoided. Thus, the average saving would be 1 day per year. Taking an industry average value of $\$0.5 \times 10^6$ per day for the cost of replacement power, the present worth of this saving over 25 years would be $\$7 \times 10^6$.

Since a major modification such as an "add-on" auxiliary feed or HPIS train would be operated in a standby role, the effect on the spurious trip role should be negligible. With care in design the additional outages for maintenance should also be negligible.

3.2.6 Presentation of the Results of the Impact Analysis

Detailed results of the estimation of the engineering costs and of the utility's additional costs are presented in tables similar to those suggested in the PNL Handbook (Tables 3.7 and 3.8 of Reference 3). An overall summary of the Impact Analysis, covering all of the Alternatives, is presented in the format shown in Table L.2. This table includes also upper and lower bounds for the avertable on site costs, to take account of the likely error in the estimates of the probability of core melt (p_m). Due to this inclusion of "Special Emergencies" as initiating events, it is considered that a factor of 5 on either side of the "Best Estimate" (or Point) value is required to provide upper and lower confidence limits.

TABLE L.2 AWO-1 - SUMMARY OF IMPACTS (Based Upon 5% Discount)

Alternative No.	Change in Core Melt Probability [Central Value]	POSITIVE IMPACTS ASSOCIATED WITH MODIFICATIONS (Present Worths)					NEGATIVE IMPACTS DUE TO AVERTABLE ONSITE COSTS (Present Worths)					NET IMPACT
		Utility Costs		Replacement Power Costs			Replace-ment Power Costs	Loss of Invest-ment Costs	Site Cleanup Costs	Total Avert-able Costs		
		Installation and Engineering Costs	Operations and Maintenance Costs (PW)	Instal-lation	In Service PW	TOTAL IMPACT						
		(\$ x10 ⁻⁶)	(\$ x10 ⁻⁶)	(\$ x10 ⁻⁶)	(\$ x10 ⁻⁶)	(\$ x10 ⁻⁶)	(\$ x10 ⁻⁶)	(\$ x10 ⁻⁶)	(\$ x10 ⁻⁶)	(\$ x10 ⁻⁶)	(\$ x10 ⁻⁶)	
j	Δp_m	I_1	13.2 I_2	I_3	13.2 I_4	TI	I'_1	I'_2	I'_3	I'_5	NI	

$$TI = I_1 + 13.2 I_2 + I_3 + 13.2 I_4$$

$$I'_5 = I'_{51} + I'_{52} + I'_{53}$$

$$NI = TI - I'_5$$

Note: In this table, L, C, and U represent the lower, central and upper bound estimates for core melt probability.

3.3 Methodology for the Value Analysis

3.3.1 The Content of the Value Analysis

The principal part of the Value Analysis is the estimation of the population dose, out to a distance of 50 miles, and the changes in dose due to the proposed Alternatives. The calculation is complicated by the fact that, for any one Alternative, the probability of a release of radioactive material in a given release category may change by a different amount from that in another category. Thus the population dose has to be estimated for each of several categories.

Other contributions to the value term are:

- a) The averted radiation exposures to plant personnel that could occur as a result of severe reactor accidents, notably the dose that would be received by those engaged in the site cleanup operations.
- b) The dose that would be received by plant and by contractor personnel in the installation phase of an Alternative. To meet NRC requirements this is treated as a negative value.
- c) The changes in occupational dose during subsequent operation, following the incorporation of an Alternative. This could be either an increase or a decrease and thus could be a negative or a positive value.

3.3.2 Estimate of Avertable Population Dose

The method for evaluating this term is as follows: for each plant the conditional population doses, out to a radius of 50 miles, have been estimated (from CRAC-2) for each release category. For each of the release categories, doses corresponding to upper bound, central value, and lower bound values for the fractional releases of individual isotopes have been calculated. The upper bound value is based on the conservatively estimated fractions in the RSS (Reference 12). The central and lower bound values are 1/3 and 1/10 of the upper bound respectively. Thus in the unmodified plant the expected population dose would be

$$\sum_{\text{All } i} p_i D_i$$

where p_i is the probability of a release in Category i , giving rise to a 50 mile dose, D_i .

Given a modification which alters the various release probabilities for individual categories, we can write the new annual expectation as

$$\sum_{\text{All } i} p_i' D_i$$

where p_i' is the new probability of release in category i . We can write:

$$\sum_{\text{All } i} p_i D_i - \sum_{\text{All } i} p_i' D_i = \sum_{\text{All } i} \delta p_i D_i$$

where $\delta p_i = p_i - p_i'$.

Thus the change in the annual expectation due to a modification, or set of modifications, can be calculated directly, given the values of δp_i and D_i .

The total avertable dose is the sum of the changes in the annual expectations over the remaining life of the plant. Thus the factor to convert annual expectation to total avertable dose over plant life is 22 for ANO-1.

The present worth in dollars of the avertable dose is obtained by multiplying the annual averted dose by \$1000 per person-rem and then discounting at 5% per year over the remaining life of the plant. The present worth factor is 13.2 for ANO-1.

3.3.3 Estimate of the Avertable On Site Dose

In the event of a severe reactor accident three possible causes of on site radiation exposure can be visualized:

- a) Dose to operating personnel on site at the time.
- b) Dose to personnel engaged in cleanup of the NPP and site.
- c) On a dual reactor site; increase in occupational dose to personnel working on the second reactor.

Each of these possible causes is discussed below.

- a) Operating personnel on site at the time of the accident.
It is considered that, with the possible exception of a few key personnel, an orderly evacuation of the site would take place prior to any release of activity. This cause would therefore not add significantly to the total exposure.

b) Dose received during cleanup operations. In the absence of better data the "rule of thumb" of 40,000 person rem suggested by NRC (Reference 2) has been adopted.

c) Additional exposure of personnel on a multi-reactor site. An upper bound to this contribution can be estimated, for a dual reactor site, on the basis that operation of the other unit would be possible, from the radiological safety point of view, when the annual dose per person would not exceed 5 rem, instead of less than an average of 1 rem per person. Assuming that a total of 200 persons would be involved, the additional dose at the ANO-1 site would be about 800 person rem per year.

Assuming that such a dose could be received following an accident (anytime in the remaining plant life) which led to core melt, but not necessarily to a large release to atmosphere, it is discounted. Thus the maximum avertable dose would not exceed 10,600 p_m person rem, i.e., substantially less than the exposure assumed for cleanup operations.

To take account of the contributions at (b) and (c) the total avertable on site dose is assumed to be 50,600 p_m person-rem. The total averted onsite dose is the sum of the annual averted onsite doses over the remaining plant life, or $50600 \Delta p_m \times 22$. In order to estimate the present worth of this term in dollars a monetary value of \$1000 per person-rem is attributed to the averted exposure and then discounted at 5% over remaining plant life of $50600 \times \$1000 \times \Delta p_m \times 13.7$.

Thus the maximum present worth of the avertable on site dose at the ANO-1 site is estimated to be $\$6.7 \times 10^8 p_m$ and the avertable cost for a change of Δp in p_m is $\$6.7 \times 10^8 \Delta p_m$.

3.3.4 Estimate of Negative Values Associated with Changes in Occupational Doses.

These are estimated by the AE (see Section 3.2.2 above) but as indicated in Section 3.2.2 above, they are treated as Values.

3.3.5 Summary of "Value" Terms

The values for each proposed Alternative are summarized in tabular form using the format illustrated in Table L.3.

The format shows, for each Alternative, the Net Values for 3 cases corresponding to the upper bound, central and lower bound release fractions defined in Section 3.3.2 above.

The format shows also the fractional reduction in population dose and in total dose (i.e., population dose plus on site doses) which would be achieved in each of the 3 cases for each Alternative.

TABLE L.3-a AWO-1 - SUMMARY OF VALUES (Based on Population Dose to 50 Miles, 5% Discount Rate)

ALTERNATIVES	POSITIVE VALUES							
	Onsite		Offsite			Total		
Change in No. Core Melt Probability (Central Value)	Averted Dose (p-rem)	Present Worth of Averted Dose @ 1000/p-rem (\$ x 10 ⁻⁶)	Averted Dose (p-rem)	Averted Dose + Base Case Dose	Present Worth of Averted Dose @ 1000/p-rem (\$ x 10 ⁻⁶)	Averted Dose (p-rem)	Averted Dose + Base Case Dose	Present Worth of Averted Dose @ 1000/p-rem (\$ x 10 ⁻⁶)
j Δp_m	V_1	V_1'	V_2	ADR _o	V_2'	V_{12}	ADR _n	V_{12}'

$$\begin{aligned}
 V_1 &= 50600 \times \Delta p_m \times 22 \\
 V_1' &= 50600 \times \Delta p_m \times \$1000 \times 13.2 \\
 V_2 &= \text{Averted Dose} \times 22 \\
 V_2' &= \text{Averted Dose} \times \$1000 \times 13.2 \\
 ADR_o &= V_2 + (\text{Basecase Dose} \times 22) \\
 V_{12} &= V_1 + V_2 \\
 ADR_n &= V_{12} + (\text{Basecase Dose} \times 22 + V_1) \\
 V_{12}' &= V_1 + V_2'
 \end{aligned}$$

Note: On this Table, L, C, and U stand for lower, central, and upper bound source term estimates.

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TABLE L.3-b AMO-1 - SUMMARY OF VALUES (Based on Population Dose to 50 Mile, 5% Discount Rate)

ALTERNATIVES	NEGATIVE VALUES						Net Value	
	Installation		Operation		Total			
Change in No. Core Melt Probability [Central Value]	Installation Dose (p-rem)	Present Worth of Installation Dose @ 1000/p-rem	In-Service Operational Dose (p-rem)	Present Worth of In-Service Oper. Dose @ (\$ x10 ⁻⁶)	Installation and Operational Dose	Present Worth of Install. & Oper. Dose @ 1000/p-rem (\$ x10 ⁻⁶)	Averted Dose (p-rem)	Present Worth of Averted Dose @ 1000/p-rem (\$ x10 ⁻⁶)
j Δp _m	V ₃	V ₃ '	V ₄	V ₄ '	V ₃ + V ₄	V ₃ ' + V ₄ '	NV	NV'

$$V_3 = V_3 \times \$1000$$

$$V_4' = (V_4 + 22) \times \$1000 \times 13.2$$

$$NV = V_1 + V_2 - V_3 - V_4$$

$$NV' = V_1' + V_2' - V_3' - V_4'$$

Note: On this Table, L, C, and U stand for lower, central, and upper bound source term estimates.

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3.3.6 Special Situations Affecting the Value Analyses

As indicated in Section 2.1 above, the NRC has emphasized the need to examine the conditions at each NPP to determine whether there are any special situations that would increase the off-site effects of a severe reactor accident.

In the case of the ANO-1 NPP, its location on the shore of Lake Dardanelle has two effects:

- a) It introduces the possibility that deposition of activity from the plume into the lake could create additional hazards due to radioactive contamination of drinking water and fish.
- b) If the population dose is based on an integration out to a limit of 50 miles, this tends to reduce the weather averaged dose by a factor of about 2 since in about one half of the possible wind directions most of the plume's travel would be over water.

A recent international collaborative study of the effects of activity deposition from reactor accidents into rivers and lakes (Reference 13) has shown that the effect of these would be small compared with the effects of deposition over land.

3.4 Methodology for the Value-Impact Analysis

Once the results of the Impact analysis and the Value analysis have been summarized on Tables L.2 and L.3, as described above, the Value-Impact analysis itself is almost self evident. For simplicity of interpretation, the results are presented in the format shown in Table L.4. This provides for the following:

- a) The analysis based on off site costs only is shown separately from the analysis based on off site and on site costs only.
- b) The results presented in Table L.4 are based on "Best Estimate" values for the various impact parameters coupled with upper, central and lower bound estimates on source terms.
- c) The predicted reduction in probability of core melt (Δp_m) and the predicted reduction in total dose (population and occupational) are shown for each Alternative. The fractional reduction in total dose is also shown.
- d) The Value-Impact relationships are shown in 3 different ways for each Alternative, namely:
 - i) Dimensionless Value/Impact ratio,
 - ii) Net benefit,

TABLE L.4 AMO-1 - SUMMARY OF VALUE-IMPACT ANALYSIS (Based on Population Dose to 50 Miles; 5% Discount Rate)

Alt. No.	Change in Core Melt Probability [Central] Value]	V-I ANALYSIS BASED ON OFFSITE COSTS										V-I ANALYSIS BASED ON OFFSITE AND ONSITE COSTS									
		Averted Offsite Dose (p-rem)	Total Averted Dose + Base Case Total Dose (U,C&L Values)	Total Impact [Central Cost] (\$ x10 ⁻⁶)	Present Worth of Averted Dose @ 1000/ p-rem [U,C&L Values] (\$ x 10 ⁻⁶)	Measures of V-I			Net Impact [Central Cost] (\$ x10 ⁻⁶)	Present Worth of Averted Dose @ 1000 person-rem [U,C&L Values] (\$ x10 ⁻⁶)	Measures of V-I										
						V-I Ratio	Net Benefit (\$ x10 ⁰)	Dollars per p-rem			V-I Ratio	Net Benefit (\$ x10 ⁰)	Dollars per p-rem								
j	Δp_j	V_2	ADR D	TI	V_2'	VIR o	NBV o	DPR o	NI	NV'	VIR n	NBV n	DPR n								

$$\begin{aligned}
 VIR_n &= V_2' + TI \\
 NBV_n &= V_2' - TI \\
 DPR_n &= TI + V_2
 \end{aligned}$$

$$\begin{aligned}
 VIR_n &= NV' + NI \\
 NBV_n &= NV' - NI \\
 DPR_n &= NI + NV
 \end{aligned}$$

iii) Dollars expended per person-rem of population dose averted.

The central estimate value-impact results are also summarized in Table L.5.

TABLE L.5 ANO-1 - SUMMARY OF VALUE-IMPACT MEASURES (Central Value)

Alter- native No.	Reduction in Core Melt Probability	Offsite Averted Dose (p-rem)	Total Averted Dose Ratio	V-I Analysis Based on Offsite Costs			V-I Analysis Based on Offsite & Onsite Costs		
				V-I Ratio	Net Benefit (\$ x 10 ⁻⁶)	Dollars per p-rem	V-I Ratio	Net Benefit (\$ x 10 ⁻⁶)	Dollars per p-rem
j	Δp_m	V_2	ADR _n	VIR _o	NBV _o	DPR _o	VIR _n	NBV _n	DPR _n

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13. ABSTRACT (200 words or less) This is one of six case studies for USI A-45 Decay Heat Removal (DHR) Requirements. The purpose of this study is to identify any potential vulnerabilities in the DHR systems of a typical Babcock and Wilcox PWR, to suggest possible modifications to improve the DHR capability, and to assess the value and impact of the most promising alternatives to the existing DHR systems. The systems analysis considered small LOCAs and transient internal initiating events, and seismic, fire, extreme wind, internal and external flood, and lightning external events. A full-scale systems analysis was performed with detailed fault trees and event trees including support system dependencies. The system analysis results were extrapolated into release categories using applicable past PRA phenomenological results and improved containment failure mode probabilities. Public consequences were estimated using site specific CRAC2 calculations. The value-impact (VI) analysis of possible alternatives considered both onsite and offsite impacts arriving at several risk measures such as averted population dose out to a 50-mile radius and dollars per person rem averted. Uncertainties in the VI analysis are discussed and the issues of feed and bleed and secondary blowdown are analyzed.			
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